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December 16, 2019

L-PI-19-031
10 CFR 50.90

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

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

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

References:

1) Letter from the Technical Specification Task Force (TSTF) to the NRC, "TSTF Comments on Draft Safety Evaluation for Traveler TSTF-505, 'Provide Risk-Informed Extended Completion Times' and Submittal of TSTF-505, Revision 2", Revision 2, dated July 2, 2018 (ADAMS Accession No. ML18183A493)

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

Pursuant to 10 CFR 50.90, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), is submitting a request for an amendment to the Technical Specifications (TS) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

The proposed amendment would modify TS requirements to permit the use of Risk Informed Completion Times in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b" (Reference 1). A model safety evaluation was provided by the NRC to the TSTF on November 21, 2018 (Reference 2).

- Attachment 1 provides a description and assessment of the proposed change, the requested confirmation of applicability, and plant-specific verifications.

- Attachment 2 provides the existing TS pages marked up to show the proposed changes.

- Attachment 3 provides existing TS Bases pages marked up to show the proposed changes and is provided for information only.

- Attachment 4 provides a cross-reference between the TS included in TSTF-505, Revision 2, and the PINGP plant-specific TS.

- Attachment 5 provides a list of implementation items that must be completed prior to implementing the Risk-Informed Completion Time Program at PINGP.

NSPM requests approval of the proposed license amendment by January 13, 2021, with an implementation period of 180 days.

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

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

Please contact Mr. Peter Gohdes at (612) 330-6503 or Peter.Gohdes@xenuclear.com if there are any questions or if additional information is needed.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

I declare under penalty of perjury, that the foregoing is true and correct.
Executed on December 16, 2019.

Scott Sharp

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (12)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ATTACHMENT 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

EVALUATION OF PROPOSED CHANGE

License Amendment Request

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

License Amendment Request

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

1.0 DESCRIPTION

The proposed amendment would modify the Technical Specification (TS) requirements related to Completion Times (CTs) for Required Actions to provide the option to calculate a longer, risk-informed CT (RICT). A new program, the Risk-Informed Completion Time Program, is added to TS Section 5, "Administrative Controls".

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

The proposed amendment is consistent with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b". However, only those Required Actions described in Attachment 4 and Enclosure 1, as reflected in the proposed TS mark-ups provided in Attachment 2, are proposed to be changed, because some of the modified Required Actions in TSTF-505 are not applicable to the Prairie Island Nuclear Generating Plant (PINGP), and there are some plant-specific Required Actions not included in TSTF-505 that are included in this proposed amendment.

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

NSPM has reviewed TSTF-505, Revision 2, and the model safety evaluation dated November 21, 2018 (Reference 1). This review included the supporting information provided to support TSTF-505 and the safety evaluation for NEI 06-09-A. As described in the subsequent paragraphs, NSPM has concluded that the technical basis is applicable to the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, and supports incorporation of this amendment in the PINGP TS.

2.2 Facility Description

NSPM owns and operates the PINGP, which is a two unit plant located on the west bank of the Mississippi River within the city limits of Red Wing, Minnesota. Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) on April 5, 1967. The Final Safety Analysis Report was submitted for application of an Operating License on January 28, 1971. Unit 1 began commercial operation on December 16, 1973, and Unit 2 began commercial operation on December 21, 1974. The PINGP Units 1 and 2 Renewed Facility Operating Licenses expire August 9, 2033, and October 29, 2034, respectively.

The PINGP was designed and constructed to comply with NSPM's understanding of the intent of the AEC 70 General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as published on July 11, 1967. PINGP was not licensed to NUREG-0800, "Standard Review Plan".

2.3 Verifications and Regulatory Commitments

2.3.1 Enclosures Provided in Accordance with NEI 06-09-A Safety Evaluation

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

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

2. Enclosure 2 provides a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models which support the RICT Program, as discussed in Regulatory Guide (RG) 1.200, Section 4.2.

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

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

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

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

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

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

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

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

11. Enclosure 11 provides a description of the implementation and monitoring program as described in NEI 06-09-A, Section 2.3.2, Step 7.

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

2.3.2 Regulatory Commitments

On May 30, 2007, the NRC issued amendments for PINGP regarding extension of the emergency diesel generator (DG) CT associated with TS 3.8.1 Required Action B.4 from 7 days to 14 days (Reference 2). As documented in Section 5.0, “Regulatory Commitments”, of the NRC safety evaluation, Nuclear Management Company (prior licensee, hereafter “NMC”), made commitments to be put in effect upon implementation of the license amendment procedures to assure that specific provisions were invoked when a DG is inoperable for beyond 7 days, described as the “extended completion time”. These provisions were the outcome of the evaluation of risk impact to address Tier 2, “Avoidance of Risk-Significant Configurations”, as defined in RG 1.177 (Reference 6). The provisions included prescribed actions to manage increases in risk due to potential combinations of equipment out of service. The NRC concluded that subsequent evaluation of proposed changes to these regulatory commitments would be best provided by the licensee's administrative processes, including its commitment management program.

NSPM will use its commitment management program to eliminate these existing prescribed actions with implementation of the RICT Program in the TS, as it meets the intent of these commitments. The Configuration Risk Management Program (CRMP) implemented with the RICT Program will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. This program is specifically designed to support the implementation of RMTS. In addition, the implementation of RMAs under the RICT Program is commensurate with the overall configuration risk significance and informed by the insights provided by the calculated component importance and important initiators for the configuration in question.

Therefore, the use of the CRMP and RMAs implemented as part of the RICT Program obviate the need for the provisions invoked in Reference 2 as the RICT Program will ensure risk impact is considered, risk-significant configurations are avoided, and adequate defense in depth is provided.

No new regulatory commitments are made in this amendment request.

2.4 Optional Variations

NSPM is proposing the following variations from the TS changes described in TSTF-505, Revision 2, or the applicable parts of the NRC staff's model safety evaluation dated November 21, 2018. These options were recognized as acceptable variations in TSTF-505 and the NRC model safety evaluation.

Note that, in a few instances, the PINGP TS utilize different numbering and titles than the NUREG-1431, "Standard Technical Specifications, Westinghouse Plants", Revision 3.1 (Reference 3), on which TSTF-505 was based. These differences are administrative and do not affect the applicability of TSTF-505 to the PINGP TS. Only TS changes consistent with the PINGP design and TS are included. Attachment 4 is a cross-reference that provides a comparison between the Required Actions included in TSTF-505 and the PINGP Required Actions included in this license amendment request. The attachment includes a summary description of the referenced Required Actions, which is provided for information purposes only and is not intended to be a verbatim description of the Required Actions. The cross-reference identifies the following:

1. PINGP Actions that have identical numbers to the corresponding NUREG-1431 Required Actions are not variations from TSTF-505, except for administrative variations (if any) such as formatting. These variations are administrative with no impact on the NRC model safety evaluation dated November 21, 2018.

2. PINGP Actions that have different numbering than the NUREG-1431 Required Actions are an administrative variation from TSTF-505 with no impact on the NRC model safety evaluation dated November 21, 2018.

3. For NUREG-1431 Required Actions that are not contained in the PINGP TS, the corresponding TSTF-505 mark-ups for the Required Actions are not applicable to PINGP. This is an administrative variation from TSTF-505 with no impact on the NRC model safety evaluation dated November 21, 2018.

4. While the TSTF-505 mark-ups were performed on Revision 3.1 of NUREG-1431, the PINGP TS are based upon Revision 1 of NUREG-1431 (Reference 4). The PINGP TS conversion to the improved TS also retained elements of the original TS that were consistent with the PINGP licensing basis and differ from NUREG-1431 Revision 1. These variations are administrative with no impact on the NRC model safety evaluation dated November 21, 2018.

5. As the proposed PINGP RICT Program is applicable in Modes 1 and 2, NSPM will not adopt changes in TSTF-505 for Required Actions that are only applicable in Mode 3 and below.

6. The model application provided in TSTF-505, Revision 2, includes an attachment for revised (clean) TS pages reflecting the proposed changes. NSPM is not including such an attachment due to the number of TS pages included in this submittal that have the

potential to be affected by other unrelated license amendment requests and the straightforward nature of the proposed changes. Providing only mark-ups of the proposed TS changes satisfies the requirements of 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit”, in that the mark-ups fully describe the changes desired. This is an administrative deviation from TSTF-505 with no impact on the NRC model safety evaluation dated November 21, 2018.

7. There are several plant-specific Limiting Conditions for Operation (LCOs) and associated Actions for which NSPM is proposing to apply the RICT Program that are variations from TSTF-505, Revision 2, as identified in Attachment 4 with additional justification provided 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.
  Condition L: One or both channel(s) inoperable on one bus.

  PINGP TS 3.3.1 Condition L is a plant-specific Condition not in the NUREG-1431 STS or TSTF-505, Revision 2. Condition L applies to the Loss of Reactor Coolant Pump through Underfrequency or Undervoltage on the 4 kV Buses (Buses 11 and 12 or 21 and 22). With one or both channel(s) inoperable on one bus, the inoperable channel(s) must be placed in trip within 6 hours. 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. A note is added to the Condition to Note that the RICT Program is only applicable when one channel on one bus is inoperable.

  The Underfrequency and Undervoltage breaker trip Functions provide protection against violating the departure from nucleate boiling ratio limit due to a loss of flow in both Reactor Coolant System (RCS) loops. Redundancy is provided by two voltage and two frequency sensors per Reactor Coolant Pump (RCP) bus. A loss of frequency or voltage detected on both RCP buses will initiate a trip of both RCP breakers. These trips will generate a reactor trip before the Reactor Coolant Flow-Low trip setpoint is reached.

  As indicated in Table E1-1 of Enclosure 1 of the PINGP TSTF-505 license amendment request (LAR), the Undervoltage breaker trip function is explicitly modeled in the PINGP PRA. The Underfrequency breaker trip function is not directly modeled. However, the Undervoltage channels are modeled, are logically equivalent, and have the same component failure rate and can be used as a surrogate for Underfrequency. The PRA Success Criterion is one of two Undervoltage channels on two of two buses.

  The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System

(RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

Therefore, the proposed revised TS 3.3.1 Condition L meets the requirements for inclusion in the RICT Program.

- TS 3.3.2 – Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO:	The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.
Condition E:	One or more Containment Pressure channel(s) inoperable.

PINGP TS 3.3.2 Condition E, which is not in-scope of TSTF-505, Revision 2, contains a plant-specific Required Action which meets the criteria for inclusion from TSTF-505, Revision 2. Condition E applies to the Containment Spray (CS) actuation on High-High Containment Pressure function. With one or more Containment Pressure channel(s) inoperable, the inoperable channel must be placed in trip and one channel verified to be OPERABLE per pair within 6 hours.

Containment pressure monitoring is accomplished through six channels derived from four pressure taps, which reflects the effectiveness of the containment and cooling systems and other engineered safety features. High pressure indicates high temperatures and reduced pressure indicates reduced temperatures. Actuation setpoints are provided in the following manner. SI actuation occurs on the lowest setpoint using a unique set of three bistables, and Steam Line isolation occurs on a higher setpoint using a unique set of three bistables. Containment Spray actuation occurs using a second setpoint on each of the SI and Steam Line Isolation bistables. The containment spray setpoint is the highest setpoint on each of the six bistables.

High-High Containment Pressure 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. Once the inoperable channel is placed in trip, the function is then such that one channel tripped on the remaining two sets will result in CS actuation. Additionally, Required Action E.1.2 requires verification that one channel per pair is OPERABLE, which precludes potential loss of function from occurring due to two channels inoperable on the same pair.

As indicated in Table E1-1 of Enclosure 1 of the PINGP TSTF-505 LAR, the CS High-High Containment Pressure function is not directly modeled in the PRA. A hydraulic analysis has been performed to show that success or failure of CS does not impact which core damage sequences are classified as contributing to LERF.

Therefore, TS 3.3.2 Condition E meets the requirements for inclusion in the RICT Program.

- TS 3.3.2 – Engineered Safety Feature Actuation System (ESFAS) Instrumentation

  LCO: The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.
  Condition I: One channel inoperable on one bus.

  PINGP TS 3.3.2 Condition I is a plant-specific Condition not in the NUREG-1431 STS or TSTF-505, Revision 2. Condition I applies to Auxiliary Feedwater (AFW)-Undervoltage on 4 kV Buses (Buses 11 and 12 or 21 and 22). With one channel inoperable on one bus, the inoperable channel must be placed in trip within 6 hours. 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. A note is added to the Condition to Note that the RICT Program is only applicable when one channel on one bus is inoperable.

  A loss of power on the buses that provide power to the Main Feedwater (MFW) pumps provides indication of a pending loss of MFW (continued) flow. The undervoltage Function senses the voltage upstream of each MFW pump breaker. Redundancy is provided by two voltage sensors per bus. A loss of power for both MFW pumps will start the turbine driven AFW pump to ensure that at least one steam generator (SG) contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

  As indicated in Table E1-1 of Enclosure 1 of the PINGP TSTF-505 LAR, the AFW-Undervoltage function is explicitly modeled in the PINGP PRA. The PRA Success Criterion is one of two channels on two of two buses.

  Therefore, TS 3.3.2 Condition I meets the requirements for inclusion in the RICT Program.

- TS 3.6.5 – Containment Spray and Cooling Systems

  LCO: Two containment spray trains and two containment cooling trains shall be OPERABLE.
  Condition C: One or both containment cooling Fan Coil Unit(s) (FCU) in one train inoperable.
  Condition D: One containment cooling FCU in each train inoperable.

  PINGP TS 3.6.5 Conditions C is based upon Condition C of the NUREG-1431 STS 3.6.6A, which is in-scope of TSTF-505, Revision 2. PINGP TS 3.6.5 Condition D is a plant specific Condition. The PINGP 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. PINGP 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 Completion Times were developed taking into account the heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a design basis accident (DBA) occurring during this period.

PINGP has two trains of containment cooling with two FCUs per train. Each train of containment cooling has sufficient capacity to supply 100% 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 PINGP TSTF-505 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 PINGP 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 containment spray can provide 100% of the required peak cooling capacity during post-accident conditions.

Therefore, TS 3.6.5 Conditions C and D meet the requirements for inclusion in the RICT Program.

- TS 3.7.1 – Main Steam Safety Valves (MSSVs)

LCO:	Five MSSVs per steam generator shall be OPERABLE.
Condition A:	One MSSV inoperable.

PINGP TS 3.7.1 Condition A, which is not in-scope of TSTF-505, Revision 2, contains a plant-specific Required Action which meets the criteria for inclusion from TSTF-505, Revision 2. Condition A is for the one MSSV inoperable with action to restore OPERABILITY of the inoperable MSSV within 4 hours. The 4 hours is a reasonable time due to the low probability of an event or transient occurring during this time requiring MSSV operation.

There are five code safety valves located on each of the two main steam lines outside the reactor containment and upstream of the main steam isolation and non-return valves. The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the RCS if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

The design basis for the MSSVs comes from American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the DBA and transient analysis. The accident analysis requires five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 100.36% reactor thermal power (RTP).

By relieving steam, the MSSVs prevent RCS overpressurization. The limiting events, described in the PINGP Updated Safety Analysis Report (USAR), that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, such as the full power turbine trip without steam dump, and increasing core heat flux events, such as the rod cluster control assembly (RCCA) withdrawal at power.

In addition to the safety valves, one power-operated relief valve is installed for each steam generator which can be manually operated from the control room. The power-operated relief valves are set to open at a pressure slightly below that of the main steam safety valves.

As indicated in Table E1-1 of Enclosure 1 of the PINGP TSTF-505 LAR, the MSSVs are explicitly modeled in the PINGP PRA. The PRA Success Criterion is one of five MSSVs per SG.

Therefore, TS 3.7.1 Condition A meets the requirements for inclusion in the RICT Program.

- TS 3.7.8 – Cooling Water (CL) System

LCO:	Two CL trains shall be OPERABLE.
Condition A:	No safeguards CL pumps OPERABLE for one train.
Condition B:	One CL supply header inoperable.

PINGP 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 the diesel driven CL pump for the train may be restored to OPERABLE status, or the 121 CL pump may be aligned to fulfill the safeguards function for the train that has no OPERABLE safeguards CL pump. The 7 day Completion Time is based on the low probability of loss of offsite power during the period; the low probability of a DBA 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. NSPM proposes 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 (MSPI).

PINGP 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 Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

The PINGP 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 (CC) System and a 100% capacity containment cooling system, has the capability to remove long term core decay heat following a design basis loss of coolant accident (LOCA) as discussed in the Section 6 of the PINGP USAR.

As indicated in Table E1-1 of Enclosure 1 of the PINGP TSTF-505 LAR, the CL System is explicitly modeled in the PINGP 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 diesel generator (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, TS 3.7.8 Conditions A and B meet the requirements for inclusion in the RICT Program.

8. Four Administrative Changes are being made to PINGP TS since the TS pages are undergoing change and review for the TSTF-505 application. The changes are as described below:

   - TS 3.3.1 revised Condition O – consistent with the remainder of the PINGP TS, a missing period is added at the end of Condition P. Although not part of TSTF-505, this change as proposed is administrative in nature as it involves a minor correction to the page to align with PINGP TS formatting.

   - TS 3.7.2 APPLICABILITY – “MODES 1” is corrected to “MODE 1”. Although not part of TSTF-505, this change as proposed is administrative in nature as it involves a minor correction to the page to align with PINGP TS formatting.

   - TS 3.7.8 Condition A – in the Notes section of the Required Action for Condition A, Note 1 is not vertically aligned with Note 2. The change will fix the alignment of Note 1 consistent with PINGP TS formatting. Although not part of TSTF-505, this change as proposed is administrative in nature as it involves a minor correction to the page to align with PINGP TS formatting.

   - TS 5.5.16, “Control Room Envelope Habitability Program” – the program heading on TS page 5.0-30 is not underlined. The change will add an underline to the heading consistent with PINGP TS formatting. Although not part of TSTF-505, this change as proposed is administrative in nature as it involves a minor correction to the page to align with PINGP TS formatting.

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

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

Application of a RICT will be evaluated using the methodology and probabilistic risk guidelines contained in NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0, which was approved by the NRC on May 17, 2007 (Reference 5). The NEI 06-09-A methodology includes a requirement to perform a quantitative assessment of the potential impact of the application of a RICT on risk, to reassess risk due to plant configuration changes, and to implement compensatory measures and risk management actions (RMAs) to maintain the risk below acceptable regulatory risk thresholds. In addition, the NEI 06-09-A methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications”, Revision 0, dated August 1998 (Reference 6), relative to the risk impact due to the application of a RICT.

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

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

NSPM has evaluated the proposed change to the TS using the criteria in 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration.

NSPM requests for PINGP, Units 1 and 2, adoption of an approved change to the standard technical specifications (STS) and plant-specific technical specifications (TS), to modify the TS requirements related to Completion Times for Required Actions to provide the option to calculate a longer, risk-informed Completion Time. The allowance is described in a new program in Chapter 5, “Administrative Controls”, entitled the “Risk-Informed Completion Time Program”.

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

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

   Response: No.

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

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

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

Response: No.

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

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

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

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NSPM concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

3.2 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

5.0 REFERENCES

1. Letter from the NRC to the Technical Specifications Task Force (TSTF), "Final Revised Model Safety Evaluation of Traveler TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4B'", dated November 21, 2018 (ADAMS Accession No. ML18269A041)

2. Letter from the NRC to NMC, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments Re: Extension of Technical Specifications 3.8.1 'AC Source – Operating,' Emergency Diesel Generator (EDG) Completion Time (TAC NOS. MC9001 and MC9002)", dated May 30, 2007 (ADAMS Accession No. ML071310023)

3. NRC NUREG-1431, Volume 1, "Standard Technical Specifications Westinghouse Plants", Revision 3.1, dated December 1, 2005

4. NRC NUREG-1431, Volume 1, "Standard Technical Specifications Westinghouse Plants", Revision 1, dated April 30, 1995 (ADAMS Accession No. ML13196A405)

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

6. NRC Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications", Revision 0, dated August 1998 (ADAMS Accession No. ML003740176)

ATTACHMENT 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

(48 Pages Follow)

INSERT EXAMPLE 1.3-8

EXAMPLE 1.3-8

ACTIONS

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

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

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

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

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

INSERT RICT

OR

In accordance with the Risk Informed Completion Time Program

INSERT RICT NOTE

-------NOTE-------
Not applicable when more than one channel inoperable on one bus.

INSERT TS 3.3.1 Condition N

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

INSERT TS 3.3.1 Condition P

P. Required Action and associated Completion Time of Condition O not met.	P.1 Reduce THERMAL POWER to < P-9.	6 hours

INSERT TS 3.3.1 Condition U

U. Required Action and associated Completion Time of Condition T not met.	U.1 Be in MODE 2.	6 hours

INSERT TS 3.3.1 Condition W

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

INSERT TS 3.3.2 Condition L

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

INSERT TS 3.3.2 Condition M

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

INSERT TS 3.3.2 Condition N

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

INSERT RICT Program

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

  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.

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.

INSERT EXAMPLE 1.3-8

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. ~~OR~~ ~~B.2 Be in MODE 3.~~	48 hours INSERT RICT ~~54 hours~~

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 INSERT RICT

ACTIONS

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
	~~OR~~	
	~~D.2 Be in MODE 3.~~	~~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~~	INSERT RICT
	~~E.2 Be in MODE 3.~~	~~12 hours~~

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
	~~OR~~	INSERT RICT
	~~K.2 Reduce THERMAL POWER to < P-7 and P-8.~~	~~12 hours~~
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
	~~OR~~	INSERT RICT NOTE INSERT RICT
	~~L.2 Reduce THERMAL POWER to < P-7 and P-8.~~	~~12 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. One Reactor Coolant Pump Breaker Open channel inoperable.	M.1 Restore channel to OPERABLE status. ~~OR~~ ~~M.2 Reduce THERMAL POWER to < P-7 and P-8.~~	48 hours INSERT RICT ~~54 hours~~
INSERT TS 3.3.1 Condition N		
~~N~~. O One Turbine Trip channel inoperable Administrative change - add period.	-----------------NOTE---------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channel(s). -- ~~N~~.1 O Place channel in trip. ~~OR~~ ~~N.2 Reduce THERMAL POWER to < P-9.~~	6 hours INSERT RICT ~~12 hours~~

INSERT TS 3.3.1 Condition P

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~O~~. 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 ~~O~~.1 Restore train to OPERABLE status.	6 hours INSERT RICT
	~~OR~~	
	~~O.2 Be in MODE 3.~~	~~12 hours~~
~~P~~. 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 ~~P~~.1 Restore train to OPERABLE status.	1 hour INSERT RICT
	~~OR~~	
	~~P.2 Be in MODE 3.~~	~~7 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~Q~~. S One or more channels inoperable. S	~~Q~~.1 Verify interlock is in required state for existing unit conditions. ~~OR~~ ~~Q.2 Be in MODE 3.~~	1 hour ~~7 hours~~
~~R~~. T One or more channels inoperable. T	~~R~~.1 Verify interlock is in required state for existing unit conditions. ~~OR~~ ~~R.2 Be in MODE 2.~~	1 hour ~~7 hours~~
INSERT TS 3.3.1 Condition U		
~~S~~. V One trip mechanism inoperable for one RTB. V	~~S~~.1 Restore ~~inoperable~~ trip mechanism to OPERABLE status. ~~OR~~ ~~S.2 Be in MODE 3.~~	48 hours INSERT RICT ~~54 hours~~

INSERT TS 3.3.1 Condition W

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	~~N~~ O	SR 3.3.1.10 SR 3.3.1.15	≥ 45 psig
b. Turbine Stop Valve Closure	$1^{(g)}$	2	~~N~~ 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	~~O~~ 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	~~Q~~ 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	~~R~~ T	SR 3.3.1.11 SR 3.3.1.13	≤ 12% RTP
2. Turbine Impulse Pressure	1	2	~~R~~ T	SR 3.3.1.7 SR 3.3.1.10	≤ 12% Full Load
c. Power Range Neutron Flux, P-8	1	4	~~R~~ T	SR 3.3.1.11 SR 3.3.1.13	≤ 11% RTP
d. Power Range Neutron Flux, P-9	1	4	~~R~~ T	SR 3.3.1.11 SR 3.3.1.13	≤ 12% RTP
e. Power Range Neutron Flux, P-10	1, 2	4	~~Q~~ S	SR 3.3.1.11 SR 3.3.1.13	≥ 9% RTP
17. Reactor Trip Breakers[h] (RTBs)	1, 2	2 trains	~~P~~ 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	~~S~~ 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	~~O~~ 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.

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. ~~OR~~ ~~B.2.1 Be in MODE 3.~~ ~~AND~~ ~~B.2.2 Be in MODE 5.~~	48 hours INSERT RICT ~~54 hours~~ ~~84 hours~~

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
		INSERT RICT
	~~OR~~	
	~~C.2.1 Be in MODE 3.~~	~~12 hours~~
	~~AND~~	
	~~C.2.2 Be in MODE 5.~~	~~42 hours~~
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
		INSERT RICT
	~~OR~~	
	~~D.2.1 Be in MODE 3.~~	~~12 hours~~
	~~AND~~	
	~~D.2.2 Be in MODE 4.~~	~~18 hours~~

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 INSERT RICT
	AND	
	E.1.2 Verify one channel per pair OPERABLE.	6 hours
	~~OR~~	
	~~E.2.1 Be in MODE 3.~~	~~12 hours~~
	~~AND~~	
	~~E.2.2 Be in MODE 4.~~	~~18 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status. ~~OR~~ ~~F.2.1 Be in MODE 3.~~ ~~AND~~ ~~F.2.2 Be in MODE 4.~~	48 hours INSERT RICT ~~54 hours~~ ~~60 hours~~
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. ~~OR~~ ~~G.2.1 Be in MODE 3.~~ ~~AND~~ ~~G.2.2 Be in MODE 4.~~	 6 hours INSERT RICT ~~12 hours~~ ~~18 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 INSERT RICT
	~~OR~~	
	~~H.2~~ ~~Be in MODE 3.~~	~~12 hours~~
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 INSERT RICT NOTE
	~~OR~~	INSERT RICT
	~~I.2~~ ~~Be in MODE 3.~~	~~12 hours~~

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

INSERT TS 3.3.2 Condition L

INSERT TS 3.3.2 Condition M

INSERT TS 3.3.2 Condition N

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE----------- Only applicable in MODE 1, 2, 3, or 4. ------------------------------- 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.	C.1 Perform SR 3.3.4.2 for OPERABLE automatic load sequencer.	6 hours <u>AND</u> Once per 24 hours thereafter
	<u>AND</u> C.2 Establish offsite paths block loading capability for associated 4 kV safeguards bus.	8 hours
	<u>AND</u> C.3 Verify offsite paths for associated 4kV safeguards bus OPERABLE.	8 hours <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> 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)
	<u>AND</u> C.5 Restore automatic load sequencer to OPERABLE status.	7 days INSERT RICT

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 INSERT RICT
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
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. <u>AND</u> B.2 Remove power from associated block valve. <u>AND</u> B.3 Restore PORV to OPERABLE status.	1 hour 1 hour 72 hours INSERT RICT
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. <u>AND</u> C.2 Restore block valve to OPERABLE status.	 1 hour 72 hours INSERT RICT

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 INSERT RICT
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
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	C.1 Enter LCO 3.0.3.	Immediately

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 INSERT RICT
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. AND 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.	4 hours INSERT RICT Once per 31 days following isolation for isolation devices outside containment AND

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. <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.	72 hours INSERT RICT 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 INSERT RICT
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
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 INSERT RICT

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One containment cooling FCU in each train inoperable.	D.1 Initiate action to isolate both inoperable FCUs. AND D.2 Restore all FCUs to OPERABLE status.	Immediately 7 days INSERT RICT
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE.

Administrative change

APPLICABILITY: MODE~~S~~ 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 INSERT RICT
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. ----------NOTE----------- Separate Condition entry is allowed for each MSIV. ---------------------------- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	8 hours Once per 7 days

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 INSERT RICT
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. OR ------------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 INSERT RICT
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 INSERT RICT

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 INSERT RICT
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

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. ~~3. This Condition may not exist > 7 days in any consecutive 30 day period.~~ --	

Administrative change - fix alignment of Note 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Restore one safeguards CL pump to OPERABLE status.	7 days INSERT RICT
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.	4 hours
	<u>AND</u>	
	B.2 Verify opposite train diesel driven CL pump OPERABLE.	4 hours
	<u>AND</u>	
	B.3 Restore CL supply header to OPERABLE status.	72 hours INSERT RICT

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.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>	
	A.2 Restore path to OPERABLE status.	7 days INSERT RICT

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One DG inoperable.	B.1 Perform SR 3.8.1.1 for the paths.	1 hour AND Once per 8 hours thereafter
	AND	
	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)
	AND	
	B.3.1 Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
	OR	
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG.	24 hours
	AND	
	B.4 Restore DG to OPERABLE status.	14 days INSERT RICT

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two paths inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features
	<u>AND</u>	
	C.2 Restore one path to OPERABLE status.	24 hours INSERT RICT
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.	12 hours INSERT RICT
	<u>OR</u>	
	D.2 Restore DG to OPERABLE status.	12 hours INSERT RICT

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
	<u>AND</u>	
	A.2 Verify the other train battery charger is OPERABLE.	2 hours
	<u>AND</u>	
	A.3 Verify the diesel generator and safeguards equipment on the other train are OPERABLE.	2 hours
	<u>AND</u>	
	A.4 Restore battery charger to OPERABLE status.	8 hours INSERT RICT

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One battery inoperable.	B.1 Verify associated battery charger is OPERABLE.	2 hours
	AND	
	B.2 Verify other train battery is OPERABLE.	2 hours
	AND	
	B.3 Verify other train battery charger is OPERABLE.	2 hours
	AND	
	B.4 Restore battery to OPERABLE status.	8 hours INSERT RICT
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 INSERT RICT
D. Required Action and Associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

3.8 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 INSERT RICT
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

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 INSERT RICT
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 INSERT RICT

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One Reactor Protection Instrument AC panel inoperable.	C.1 Restore Reactor Protection Instrument AC panel to OPERABLE status.	2 hours INSERT RICT
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
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

Administrative change - underline

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.

INSERT RICT Program

ATTACHMENT 3

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

PROPOSED CHANGES TO TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP) PAGES (PROVIDED FOR INFORMATION ONLY)

(54 Pages Follow)

INSERT TSB RICT 1

or in accordance with the Risk Informed Completion Time Program.

INSERT TSB RICT 2

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

INSERT TSB 3.3.1 Condition N

N.1

If the Required Action and associated Completion Time 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. This places the unit in a MODE where the LCO is no longer applicable.

INSERT TSB 3.3.1 Condition P

P.1

If the Required Action and associated Completion Time of Condition O is not met, THERMAL POWER must be reduced below the P-9 setpoint within 6 hours. This places the unit in a MODE where the LCO is no longer applicable.

INSERT TSB 3.3.1 Condition U

U.1

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. The Completion Time 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.

INSERT TSB 3.3.1 Condition W

W.1

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. The Completion Time 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. 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.

INSERT TSB 3.3.2 Condition L

L.1

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

INSERT TSB 3.3.2 Condition M

M.1

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

INSERT TSB 3.3.2 Condition N

N.1

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. 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.

INSERT TSB RICT NOTE

This Condition has been modified by a NOTE to require that application of the Risk Informed Completion Time Program is not applicable when more than one channel on one bus is inoperable. The PRA Success Criterion is one of two channels on two of two buses. As previously described in the Bases, with one or both channel(s) per bus inoperable, action must be taken to place the inoperable channel(s) in trip.

BASES

ACTIONS (continued)

Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified (e.g., on a per steamline, per loop, per SG, etc., basis), then the Condition may be entered separately for each steamline, loop, SG, etc., as applicable.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit may be outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 ~~and B.2~~

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the Reactor Protection Relay Logic for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours~~.~~ INSERT TSB RICT 1 In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

BASES

ACTIONS

B.1 ~~and B.2 (continued)~~

~~If the channel cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time). The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, Action C would apply 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.~~

C.1 , C.2.1 and C.2.2

Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or one or more rods not fully inserted:

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the RTS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, action must be initiated within the

BASES

ACTIONS

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

same 48 hours to ensure that all rods are fully inserted and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With rods fully inserted and the Rod Control System incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

and

D.1.1~~,~~ D.1.2~~, and D.2~~

Condition D applies to the following reactor trip Functions:

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

The NIS power range detectors provide input to the reactor control system and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 6).

INSERT
TSB RICT 2

BASES

ACTIONS

D.1.1~~,~~ and D.1.2~~, and D.2~~ (continued)

In addition to placing the inoperable channel in the tripped condition, monitor the QPTR once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels > 85% RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

~~As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.~~

If Condition D is entered while performing PHYSICS TESTS in accordance with LCO 3.1.8, a total of two channels may be inoperable.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 6.

Required Action D.1.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if THERMAL POWER is > 85% RTP and the Power Range Neutron Flux input to QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function

BASES

and

ACTIONS

D.1.1~~,~~ D.1.2~~, and D.2~~ (continued)

inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the core power distribution measurement information once per 12 hours may not be necessary.

E.1 ~~and E.2~~

Condition E applies to the following reactor trip Functions:

- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-High; and
- SG Water Level-Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours~~.~~ Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 6.

INSERT TSB RICT 1

~~If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.~~

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up

BASES

ACTIONS

J.1 and J.2 (continued)

OPERABLE status, action must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the Rod Control System must be placed in a condition incapable of rod withdrawal within the next hour. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour are justified in Reference 6.

K.1 ~~and K.2~~

Condition ~~K~~ applies to the following reactor trip Functions:

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

INSERT TSB RICT 1

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours~~.~~ Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 or P-8 setpoints. These Functions do not have to be OPERABLE below the P-7 and P-8 setpoints because there are no loss of flow trips below these setpoints. There is insufficient heat production to generate DNB conditions below these setpoints. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 6. ~~An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 and P-8 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.~~

BASES

ACTIONS

K.1 ~~and K.2~~ (continued)

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channels, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

L.1 ~~and L.2~~

Condition L applies to the Loss of Reactor Coolant Pump Underfrequency 4 kV Buses 11 and 12 (21 and 22) and Undervoltage on 4 kV Buses 11 and 12 (21 and 22). With one ~~or both~~ channel~~s~~ inoperable on one bus, the inoperable channel~~(s)~~ must be placed in trip within 6 hours~~.~~ INSERT TSB RICT 1 ~~If the channel(s) cannot be restored to OPERABLE status or the channel(s) placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-7 and P-8 setpoints within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable.~~ These trip Functions do not have to be OPERABLE below the P-7 and P-8 setpoints because analyses demonstrate AOOs meet their DNB criteria without requiring these trip functions at this low power level. The 6 hours allowed to restore the channel~~(s)~~ to OPERABLE status ~~or place in trip and the 6 additional hours allowed to reduce THERMAL POWER to below the P-7 and P-8 setpoints are~~ is justified in Reference 6.

The Required Actions have been modified by a Note that allows placing one inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 6.

INSERT TSB RICT NOTE

BASES

ACTIONS (continued)

M.1 ~~and M.2~~

Condition M applies to the RCP Breaker Open reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours~~.~~ INSERT TSB RICT 1 The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains, other breaker position channels, other flow related trip Functions and the low probability of an event occurring during this interval.

~~If the channel cannot be restored to OPERABLE status within the 48 hours, then THERMAL POWER must be reduced below the P-7 and P-8 setpoints within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable.~~ This Function does not have to be OPERABLE below the P-7 and P-8 setpoints because analyses demonstrate AOOs meet their DNB criteria without requiring this Trip Function at this low power level.

INSERT TSB 3.3.1 Condition N

O ~~N~~.1 ~~and N.2~~

Condition O ~~N~~ applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours~~.~~ INSERT TSB RICT 1 If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. ~~If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 6 hours.~~ The 6 hours allowed to place the inoperable channel in the tripped condition and the 6 hours allowed for reducing power are justified in Reference 6.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channel(s). The 4 hour time limit is justified in Reference 6.

INSERT TSB 3.3.1 Condition P

BASES

ACTIONS
(continued)

~~O~~ Q .1 ~~and O.2~~

Condition ~~O~~ Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status ~~(Required Action O.1) or the unit must be placed in MODE 3 within the next 6 hours~~. The Completion Time of 6 hours (Required Action ~~O~~ Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. ~~The Completion Time of an additional 6 hours (Required Action O.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.~~

INSERT TSB RICT 2

The Required Actions have been modified by a Note that allows bypassing one train up to 8 hours for surveillance testing, provided the other train is OPERABLE. A train is normally bypassed by placing the bypass breaker in service and opening the associated RTB. The RTB remains OPERABLE under these conditions so that entry into Condition P is not required while performing testing allowed by this Note.

~~P~~ R .1 ~~and P.2~~

Condition ~~P~~ R applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one RTB train inoperable, 1 hour is allowed to restore the train to OPERABLE status ~~or the unit must be placed in MODE 3 within the next 6 hours~~. ~~The Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 7 hour Completion Times are equal to the~~

INSERT TSB RICT 2

BASES

ACTIONS

~~P~~R.1~~ and P.2~~ (continued)

~~time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 results in Action C entry while RTB(s) are inoperable.~~

When the Automatic Relay Logic train associated with a RTB train is inoperable and Condition ~~O~~Q has been entered, the RTB is normally bypassed by placing the bypass breaker in service and opening the associated RTB. The RTB remains OPERABLE under these conditions so that entry into Condition ~~P~~S is not required.

The Required Actions have been modified by two Notes. Note 1 allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 4 hours for maintenance on undervoltage or shunt trip mechanisms if the other train is OPERABLE.

~~Q~~S.1~~ and Q.2~~

Condition ~~Q~~S applies to the P-6 and P-10 interlocks. With one or more channel(s) inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour~~ or the unit must be placed in MODE 3 within the next 6 hours~~. Verifying the interlock status ensures the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions.~~ The Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 7 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.~~

BASES

ACTIONS (Continued)

~~R~~.1 ~~and R.2~~ T T

Condition ~~R~~ T applies to the P-7, P-8, and P-9 interlocks. With one or more channel(s) inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour ~~or the unit must be placed in MODE 2 within the next 6 hours~~. These actions are conservative for the case where power level is being raised. Verifying the interlock status ensures the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. ~~The Completion Time of an additional 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit~~ systems.

INSERT TSB 3.3.1 Condition U

~~S~~.1 ~~and S.2~~ V V

Condition ~~S~~ V applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours ~~or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of an additional 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.~~

INSERT TSB RICT 1

~~With the unit in MODE 3, Action C would apply to any inoperable RTB Trip mechanism.~~ The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 4 hours, per Condition ~~P~~ R.

BASES

ACTIONS

V ~~S~~.1 ~~and S.2~~ (continued)

The Completion Time of 48 hours for Required Action V ~~S~~.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

INSERT TSB 3.3.1 Condition W

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of reactor protection analog system supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect

BASES

ACTIONS

A.1 (continued)

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1~~, B.2.1, and B.2.2~~

Condition B applies to manual initiation of:

- SI;
- Containment Spray (CS); and
- Containment Isolation (CI).

INSERT TSB RICT 2

This action addresses the train orientation of the ESF relay logic for the functions listed above. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each Function (except for CS), and the low probability of an event occurring during this interval. ~~If the channel cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

BASES

ACTIONS (continued)

C.1~~, C.2.1, and C.2.2~~

Condition C applies to the automatic actuation relay logic for the following functions:

- SI;
- CS; and
- CI

INSERT TSB RICT 2

This action addresses the train orientation of the ESF relay logic. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. ~~If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

The Required Actions are modified by a Note that allows one train to be bypassed for up to 8 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 5) that 8 hours is the average time required to perform relay logic train surveillance.

BASES

ACTIONS (continued)

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

Condition D applies to:

- High Containment Pressure;
- Pressurizer Low Pressure;
- Steam Line Low Pressure;
- Steam Line Isolation High High Containment Pressure ;
- High Steam Flow Coincident With Safety Injection Coincident With Low Low T_{avg};
- High High Steam Flow Coincident With Safety Injection; and
- Low Low SG Water Level.

INSERT TSB RICT 2

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-three configuration that satisfies redundancy requirements.

~~Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.~~

BASES

ACTIONS

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

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

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 5.

and

E.1.1~~,~~ E.1.2~~, E.2.1, and E.2.2~~

Condition E applies to CS High High Containment Pressure which is a one-out-of-two channels, three-out-of-three sets logic. Condition E addresses the situation where containment pressure channels are inoperable. With channel(s) tripped, one or more of the three sets may be actuated.

INSERT TSB RICT 2

Restoring the channel to OPERABLE status, or placing the other inoperable channel in the trip condition and verifying one channel in each pair remains OPERABLE within 6 hours, is sufficient to assure that the Function remains OPERABLE. The Completion Time is further justified based on the low probability of an event occurring during this interval. ~~Failure to restore the inoperable channel(s) to OPERABLE status, or place it in the tripped condition within 6 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed~~

BASES

ACTIONS

and

E.1.1~~,~~ E.1.2~~, E.2.1, and E.2.2~~ (continued)

~~Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, this Function is a no longer required OPERABLE.~~

The Required Actions are modified by a Note that allows one channel to be bypassed for up to 4 hours for surveillance testing. Placing a channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference 5.

F.1~~, F.2.1, and F.2.2~~

INSERT
TSB RICT 2

Condition F applies to Manual Initiation of Steam Line Isolation. If a train or channel is inoperable, 48 hours are allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of this Function and the low probability of an event occurring during this interval. ~~If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.~~

BASES

ACTIONS (continued)

G.1~~, G.2.1, and G.2.2~~

INSERT TSB RICT 2

Condition G applies to the automatic actuation relay logic for the Steam Line Isolation and Feedwater Isolation Functions. The action addresses the train orientation of the ESF relay logic for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. ~~If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the actuation function. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the Functions noted above.~~

The Required Actions are modified by a Note that allows one train to be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 5) assumption that 8 hours is the average time required to perform relay logic train surveillance.

INSERT TSB RICT 2

H.1~~ and H.2~~

Condition H applies to High High SG Water Level.

If one channel is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two logic will result in actuation.

BASES

ACTIONS

H.1 ~~and H.2~~ (continued)

The 6 hour Completion Time is justified in Reference 5. ~~Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. 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, this Function is no longer required OPERABLE.~~

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 5.

I.1 ~~and I.2~~

Condition I applies to Undervoltage on Buses 11 and 12 (21 and 22).

If one or both channel(s) on one bus is inoperable, 6 hours are allowed to restore the channel(s) to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two channels on the other bus will result in actuation. The 6 hour Completion Time is justified in Reference 5. ~~Failure to restore the inoperable channel(s) to OPERABLE status or place it in the tripped~~

INSERT TSB RICT 2

BASES

ACTIONS

I.1 ~~and I.2~~ (continued)

~~condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. 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, this Function is no longer required OPERABLE.~~

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 5.

INSERT TSB RICT NOTE

J.1 and K.1

Conditions J and K apply to the AFW automatic actuation relay logic function and to the AFW pump start on trip of both MFW pumps function.

The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a logic train or channel is inoperable, the applicable Condition(s) and Required Action(s) of LCO 3.7.5, "Auxiliary Feedwater (AFW) System," are entered for the associated AFW Train or pump.

Required Action J.1 is modified by a note that allows placing a train in the bypass condition for up to 8 hours for surveillance testing provided the other train is OPERABLE. This is necessary to allow testing reactor trip system logic which is in the same cabinet with AFW logic. This is acceptable since the other AFW system train is OPERABLE and the probability for an event requiring AFW during this time is low.

INSERT TSB 3.3.2 Condition M

INSERT TSB 3.3.2 Condition N

INSERT TSB 3.3.2 Condition O

BASES

ACTIONS

C.4 (continued)

however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

C.5

INSERT
TSB RICT 2

Required Action C.5 requires that the automatic load sequencer be restored to OPERABLE status. The 7 day Completion Time allows a reasonable time to repair the inoperable load sequencer. The Completion Time is consistent with the Completion Time to restore an inoperable DG, as required in LCO 3.8.1, "AC Sources - Operating."

D.1

Condition D applies when the Required Action and associated Completion Time of Condition C are not met. The unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours.

E.1

Required Action E.1 requires that LCO 3.8.2 "AC Sources-Shutdown" Condition(s) and Required Action(s) for the DG made inoperable from inoperable 4 kV safeguards bus voltage

BASES

APPLICABILITY (continued)

for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Allowable Value for Pressurizer High Water Level-Reactor Trip.

If the pressurizer water level is not within the limit, action must be taken to bring the unit to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3, with all rods fully inserted and incapable of withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

INSERT
TSB RICT 1

B.1

If one group of pressurizer heaters is inoperable, restoration is required within 72 hours~~.~~ The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

BASES

ACTIONS (continued)

B.1, B.2 and B.3

If one PORV is inoperable for reasons other than Condition A, and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on the small potential for challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

INSERT
TSB RICT 2

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of

INSERT
TSB RICT 2

BASES

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA and meet required parameters for mitigation of a secondary side loss of fluid accident. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above.

In MODES 4, 5, and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. MODE 4 core cooling requirements are addressed by LCO 3.5.3, "ECCS-Shutdown," and LCO 3.4.6, "RCS Loops-MODE 4." Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

<u>A.1</u>

INSERT
TSB RICT 1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours~~.~~ The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 4) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or required supporting systems are not available.

BASES

ACTIONS

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

test or if the overall air lock leakage is not within the limits of SR 3.6.2.1. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits due to the large margin between the air lock leakage and the containment overall leakage acceptance criteria.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time~~.~~ The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

INSERT
TSB RICT 1

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

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

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

<u>A.1 and A.2</u>

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

INSERT TSB RICT 1

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no

BASES

ACTIONS

A.1 and A.2 (continued)

following isolation

longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

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

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

BASES

ACTIONS (continued)

C.1 and C.2

INSERT TSB RICT 1

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated power operated valve, a closed manual valve, and a blind flange. With the exception of the chemical and volume control system (CVCS), a check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time~~.~~ The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. This required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

following isolation

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements defined in Reference 2. This Note is

BASES

LCO (continued)

up to and including 95ºF. If Technical Specification (TS) 3.6.5 Condition D has been entered and the CL supply temperature does not exceed 95ºF, then the remaining two containment cooling fan coil units provide adequate heat removal within the TS 3.6.5 Condition D allowed Completion Time.

APPLICABILITY

In MODES 1, 2, 3, and 4, a LOCA or SLB could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

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

ACTIONS

A.1

INSERT
TSB RICT 1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours~~.~~ In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the other Containment Spray train, reasonable time for repairs, and low probability of a LOCA or SLB occurring during this period.

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

INSERT
TSB RICT 1

With one or both of the containment cooling fan coil units (FCU) in one train inoperable, the inoperable FCU(s) must be restored to OPERABLE status within 7 days~~.~~ In this degraded condition the remaining OPERABLE containment spray and cooling trains provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

D.1 and D.2

Condition D applies when one FCU in each train is inoperable. With two FCUs inoperable, the Required Actions are to isolate cooling water flow to both inoperable FCUs immediately. This will assure the containment cooling function continues to be provided.

The LCO requires the OPERABILITY of a number of components within the subsystems. Due to the redundancy of components within the containment cooling system, the inoperability of two FCU does not render the containment cooling system incapable of performing its function. Engineering analyses demonstrate that two

BASES

ACTIONS

D.1 and D.2 (continued)

OPERABLE FCUs, one in each train, are capable of providing the necessary cooling.

With a FCU inoperable in both containment cooling trains and a FCU OPERABLE in both containment cooling trains, the two remaining OPERABLE FCUs can provide the necessary cooling provided the cooling water flow to the inoperable FCUs is isolated.

When one FCU in each containment cooling train is inoperable, both inoperable FCUs must be restored to OPERABLE status within 7 days~~.~~ In this degraded condition the remaining OPERABLE containment spray and FCU from each cooling train provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

INSERT TSB RICT 1

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4, 5, and 6, there are no credible transients requiring the MSSVs.

The energy content in the steam generators is sufficiently low in MODES 5 and 6 that they cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

A.1

INSERT
TSB RICT 1

With one MSSV inoperable, restore OPERABILITY of the inoperable MSSV within 4 hours~~.~~ The 4 hours is a reasonable time due to the low probability of an event or transient occurring during this time requiring MSSV operation.

Continued operation with less than all five MSSVs OPERABLE for each steam generator is not permitted since safety analyses supporting such operation have not been performed.

B.1 and B.2

If the MSSV cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within

BASES (continued)

LCO

This LCO requires that both MSIVs be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on a main steam isolation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally the MSIVs are closed, and the steam generator energy is low. In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

INSERT TSB RICT 1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours~~.~~ Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs and considering the redundancy of the NRCV.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional passive means for containment isolation.

BASES (continued)

ACTIONS

A.1

INSERT TSB RICT 1

With one required SG PORV line inoperable, action must be taken to restore OPERABLE status within 7 days~~.~~

The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE SG PORV lines, Steam Dump System, and MSSVs.

B.1

With two SG PORV lines inoperable, action must be taken to restore one SG PORV to OPERABLE status. Since the block valve can be closed to isolate a SG PORV, some repairs may be possible with the unit at power.

The 1 hour Completion Time allows time to plan an orderly shutdown of the unit and is reasonable, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the SG PORV lines.

C.1 and C.2

If the SG PORV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 12 hours.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to provide heat removal. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required to perform a safety function.

ACTIONS

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

INSERT
TSB RICT 1

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, or if a turbine driven pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days~~.~~ The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump;

BASES

ACTIONS

A.1 (continued)

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling outage, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation; and

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of the redundant OPERABLE motor driven AFW pump, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

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

B.1

INSERT TSB RICT 1

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

BASES

ACTIONS

A.1 (continued)

"RCS Loops-MODE 4," be entered if an inoperable CC train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

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

INSERT
TSB RICT 1

B.1 and B.2

If the CC train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CC flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CC System.

BASES

LCO
(continued)

b. If the piping or component inoperability results in required components in a train being incapable of heat removal, the train is to be considered inoperable; and

c. If cooling flow for the required components can be maintained by opening the emergency dump to grade path, by routing to the other unit's discharge header, or overflow from the turbine building standpipes, the train or components are not considered inoperable.

APPLICABILITY

The CL System specification is applicable for single or two unit operation.

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

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

ACTIONS

A.1

INSERT
TSB RICT 1

If no safeguards CL pumps are OPERABLE for one train, action must be taken to restore one CL safeguards pump to OPERABLE status within 7 days~~.~~

Either the diesel driven CL pump for the train may be restored to OPERABLE status, or the 121 CL pump may be aligned to fulfill the safeguards function for the train that has no OPERABLE safeguards CL pump.

The 7 day Completion Time is based on:

a. Low probability of loss of offsite power during the period;

BASES

ACTIONS

A.1 (continued)

b. The low probability of a DBA occurring during this time period;

c. The safeguards cooling capabilities afforded by the remaining OPERABLE train; and

d. The capability to route water from the non-safeguards pumps, if needed.

Required Action A.1 is modified by 3 Notes. Note 1 requires Unit 1 entry into the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," for an emergency diesel generator made inoperable by the CL System. For Unit 1, the diesel generators are major heat loads supplied by the CL System. Thus, inoperability of two safeguards CL pumps will affect at least the heat loads on one CL header, including one Unit 1 diesel generator. Inability to adequately remove the heat from the diesel generator will render it inoperable.

Note 2 requires entry into the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4", for both units for the RHR loops made inoperable by the CL System. If either unit is in MODE 4, inoperability of two safeguards CL pumps may affect all the heat loads on one CL header, including a CC train and subsequently one RHR heat exchanger on each unit. Inability to adequately remove the heat from a RHR heat exchanger will render it inoperable.

~~Note 3 specifies that the Condition with no safeguard CL pumps OPERABLE for one train may not exist for more than 7 days in any consecutive 30 day period. If such a condition occurs, Condition C must be entered with the specified Required Action taken because the equipment reliability is less than considered acceptable.~~

BASES

ACTIONS (continued)

B.1, B.2 and B.3

INSERT TSB RICT 1

If one CL supply header is inoperable, action must be taken to verify the vertical motor driven CL pump and the opposite train diesel driven CL pump are OPERABLE within 4 hours, and restore the inoperable CL header to OPERABLE status within 72 hours~~.~~

Verification of vertical motor driven CL pump OPERABILITY does not require the pump to be aligned and may be performed by administrative means. Verification of the opposite train diesel driven CL pump may be performed by administrative means. Completion of the CL pump surveillance tests is not required.

Conditions may occur in the CL System piping, valves, or instrumentation downstream of the supply header (e.g., closed or failed valves, failed piping, or instrumentation in a return header) that can result in the supply header being considered inoperable. In such cases, Condition B and related Required Actions shall apply.

In this Condition, the remaining OPERABLE CL header is adequate to perform the heat removal function. However, the overall redundancy is reduced because only a single CL train remains OPERABLE.

Required Action B.1 ensures that the vertical motor driven 121 CL pump may be used to provide redundancy for the safeguards CL pump on the OPERABLE header. Required Action B.3 assures adequate system reliability is maintained.

Required Actions B.1, B.2, and B.3 are modified by two Notes.

The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources-Operating," should be entered for Unit 1 since an inoperable CL train results in an inoperable emergency diesel generator.

BASES

ACTIONS (continued)

A.2

INSERT TSB RICT 2

Operation may continue in Condition A for a period that should not exceed 7 days. With one path inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE path and DGs are adequate to supply electrical power to the onsite Safeguards Distribution System.

The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the paths on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a path fails to pass SR 3.8.1.1, it is inoperable and additional Conditions and Required Actions apply.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

BASES

ACTIONS (continued)

B.3.1 and B.3.2

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

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

According to the Maintenance Rule, 24 hours is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG.

B.4

Operation may continue in Condition B for a period that should not exceed 14 days.

In Condition B, the remaining OPERABLE DG and paths are adequate to supply electrical power to the onsite Safeguards Distribution System. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

INSERT TSB RICT 2

BASES

ACTIONS (continued)

C.1 and C.2

Required Action C.1, which applies when two paths are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is 12 hours. The rationale for the 12 hours is that a Completion Time of 24 hours is allowed for two paths inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains.

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

a. Both paths are inoperable; and

b. A required feature on either train is inoperable.

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

INSERT TSB RICT 2

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

BASES

ACTIONS

C.1 and C.2 (continued)

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

With the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two paths are restored within 24 hours, unrestricted operation may continue. If only one path is restored within 24 hours, power operation continues in accordance with Condition A.

D.1 and D.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to either train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems-Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one path and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

Operation may continue in Condition D for a period that should not exceed 12 hours. INSERT TSB RICT 2

In Condition D, redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since

BASES

ACTIONS

D.1 and D.2 (continued)

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

INSERT
TSB RICT 2

E.1

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

With both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

BASES (continued)

ACTIONS

A.1, A.2, A.3, and A.4

Condition A represents one battery charger inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). Required Actions A.1 and A.2 verify that the associated battery and other train charger are OPERABLE within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or verifying that the associated battery and other train charger are OPERABLE and no loss of function exists.

Required Action A.3 requires, within 2 hours, that the diesel generator and safeguards equipment on the other train are verified to be OPERABLE. This verification ensures that the redundant train is OPERABLE ensuring that the plant will be able to mitigate an event as analyzed in the USAR (Ref. 3).

Required Action A.4 limits the restoration time for the inoperable battery charger to 8 hours. The 8 hour Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

INSERT
TSB RICT 2

B.1, B.2, B.3, and B.4

Condition B represents one battery inoperable. With one battery inoperable, the DC bus is being supplied by the OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in loss of DC to that train. Recovery of the AC bus, especially if it is due to a loss of offsite power, will be hampered by the fact that many of the components necessary for the recovery (e.g., diesel generator control and field flash, AC load shed and diesel generator output circuit breakers, etc.) likely rely upon the battery. Required Actions B.1, B.2, and B.3

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

verify that the associated battery charger, the other train battery and associated charger are OPERABLE within 2 hours. This time provides for either returning the inoperable battery to OPERABLE status or verifying that the associated charger and other train battery and charger are OPERABLE therefore, ensuring no loss of function exists.

INSERT TSB RICT 2

Required Action B.4 requires the inoperable battery to be restored to OPERABLE within 8 hours. The 8 hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery inoperability (e.g., loss of battery charger, battery cell voltage less than 2.07 V, etc.) are identified in Specifications 3.8.4, 3.8.5, and 3.8.6 together with additional specific completion times.

C.1

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

INSERT TSB RICT 2

If one of the required DC electrical power subsystems is inoperable for reasons other than Condition A or B (e.g., inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of minimum necessary DC electrical subsystems to mitigate a worst

BASES (continued)

ACTIONS

A.1

With one required Reactor Protection Instrument AC inverter inoperable, its associated Reactor Protection Instrument AC panel becomes inoperable until it is re-energized from an operable inverter or the inverter internal bypass source.

For this reason a Note has been included in Condition A requiring entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems – Operating." This ensures that the Reactor Protection Instrument AC panel is re-energized within 2 hours.

Required Action A.1 allows 24 hours to restore the inoperable Reactor Protection Instrument AC inverter to OPERABLE status. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the Reactor Protection Instrument AC panel is powered form its alternate source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the Reactor Protection Instrument AC panel is the preferred source for powering instrumentation trip setpoint devices.

INSERT
TSB RICT 2

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

APPLICABILITY (continued)

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

ACTIONS

A.1

With one or more safeguards AC electrical power distribution subsystems, inoperable, the remaining AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, required safeguards AC electrical power, distribution subsystems to be restored to OPERABLE status within 8 hours~~.~~ INSERT TSB RICT 1

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

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

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

BASES

ACTIONS

A.1 (continued)

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

B.1

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

INSERT
TSB RICT 2

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

BASES

ACTIONS

B.1 (continued)

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

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

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

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

C.1

With one Reactor Protection Instrument AC panel inoperable, the remaining OPERABLE Reactor Protection Instrument AC panels are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum ESF functions not being supported. Therefore, the required Reactor Protection Instrument AC panel must be restored to OPERABLE status within 2 hours by powering the panel from the associated inverter or inverter bypass transformer.

INSERT
TSB RICT 2

Condition C represents one Reactor Protection Instrument AC panel without power. In this situation, the unit is significantly more

ATTACHMENT 4

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

CROSS REFERENCE OF TSTF-505 AND PINGP TECHNICAL SPECIFICATIONS (PROVIDED FOR INFORMATION ONLY)

(14 Pages Follow)

Cross-Reference of TSTF-505 and PINGP Technical Specifications

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Completion Times	**1.3**	**1.3**		
Example 1.3-8	[NEW TS] 1.3-8	[NEW TS] 1.3-8	No	The PINGP TS do not currently contain this example. Example to be added to the TS to be consistent with TSTF-505. This is a new definition only (i.e., there is no RICT directly applicable to the TS).
Reactor Trip System (RTS) Instrumentation	**3.3.1**	**3.3.1**		
One Manual Reactor Trip channel inoperable.	3.3.1.B.1	3.3.1.B.1	Yes	TSTF-505 changes are incorporated.
One channel or train inoperable.	3.3.1.C.1	3.3.1.C.1	No	The proposed PINGP RICT Program is applicable in Modes 1 and 2. This Condition is applicable in Modes 3, 4, and 5. Therefore, TSTF-505 changes are not incorporated.
One Power Range Neutron Flux – High channel inoperable.	3.3.1.D.1.1 3.3.1.D.2.1	- 3.3.1.D.1.1	Yes	PINGP TS Condition D is “One Power Range Neutron Flux channel inoperable”. PINGP TS do not have an equivalent to STS 3.3.1.D.1.1. TSTF-505 changes are incorporated for the Required Actions applicable to the PINGP TS.
One channel inoperable.	3.3.1.E.1	3.3.1.E.1	Yes	TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
One Source Range Neutron Flux channel inoperable.	3.3.1.J.1	3.3.1.J.1	No	The proposed PINGP RICT Program is applicable in Modes 1 and 2. This Condition is applicable in Modes 3, 4, and 5. Therefore, TSTF-505 changes are not incorporated.
Required Action and associated Completion Time of Condition C or J not met.	[NEW] 3.3.1.K	-	No	The PINGP TS do not currently contain this Condition. As RICTs are not being incorporated into TS 3.3.1.C or TS 3.3.1.J, this new Condition will not be added to the PINGP TS.
One channel inoperable.	3.3.1.L.1	3.3.1.K.1	Yes	TSTF-505 changes are incorporated.
[PINGP TS Condition] One or both channel(s) inoperable on one bus.	-	3.3.1.L.1	Yes	This is a PINGP-specific Condition. Both channels inoperable is a loss of function, therefore, NSPM proposes adding a note to limit applicability of a RICT to one channel inoperable. Therefore, changes consistent with TSTF-505 are incorporated.
Required Action and associated Completion Time of Condition L not met.	3.3.1.M.1	-	No	See new PINGP TS 3.3.1.N.
One Reactor Coolant Pump Breaker Position (Single Loop) channel inoperable.	3.3.1.N.1	3.3.1.M.1	Yes	PINGP TS Condition M is “One Reactor Coolant Pump Breaker Open channel inoperable”, which is consistent with NUREG-1431, Revision 1. TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Required Action and associated Completion Time of Condition N not met.	[NEW] 3.3.1.M 3.3.1.O 3.3.1.Q	[NEW] 3.3.1.N	No	The PINGP TS do not currently contain this Condition. This new Condition will be added consistent with TSTF-505. However, at PINGP, P-7 and P-8 are approximately the same power level, such that a power operating region of >P-7 and <P-8 does not exist. Also, existing PINGP TS 3.3.1 Required Actions K.2, L.2, and M.2 all allow an additional 6 hours before reducing thermal power. Therefore, new TSTF-505 Conditions M, O, and Q can be consolidated into one new PINGP TS Condition N.
One Reactor Coolant Breaker Position (Two Loops) channel inoperable.	3.3.1.P.1	-	No	The PINGP TS do not contain this Condition. Therefore, TSTF-505 changes are not incorporated.
Required Action and associated Completion Time of Condition P not met.	3.3.1.Q.1	-	No	See PINGP TS 3.3.1.N.
One Turbine Trip channel inoperable.	3.3.1.R.1	3.3.1.O.1	Yes	TSTF-505 changes are incorporated.
Required Action and associated Completion Time of Condition R not met.	[NEW] 3.3.1.S	[NEW] 3.3.1.P	No	The PINGP TS do not currently contain this Condition. This new Condition will be added based on TSTF-505 and consistent with the existing PINGP TS.
One train inoperable.	3.3.1.T.1	3.3.1.Q.1	Yes	TSTF-505 changes are incorporated.
One RTB train inoperable.	3.3.1.U.1	3.3.1.R.1	Yes	TSTF-505 changes are incorporated.
One or more channels inoperable.	3.3.1.V.1	3.3.1.S.1	No	TSTF-505 changes are incorporated (not in RICT scope).
One or more channels inoperable.	3.3.1.W.1	3.3.1.T.1	No	TSTF-505 changes are incorporated (not in RICT scope).

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Required Action and associated Completion Time of Condition W not met.	3.3.1.X	3.3.1.U	No	The PINGP TS do not currently contain this Condition. This new Condition will be added consistent with TSTF-505.
One trip mechanism inoperable for one RTB.	3.3.1.Y.1	3.3.1.V.1	Yes	TSTF-505 changes are incorporated.
Required Action and associated Completion Time of Condition B, D, E, T, U, V, W, or Y not met.	3.3.1.Z	3.3.1.W	No	The PINGP TS do not currently contain this Condition. This new Condition will be added consistent with TSTF-505.
Engineered Safety Feature Actuation System (ESFAS) Instrumentation	**3.3.2**	**3.3.2**		
One channel or train inoperable.	3.3.2.B.1	3.3.2.B.1	Yes	TSTF-505 changes are incorporated.
One train inoperable.	3.3.2.C.1	3.3.2.C.1	Yes	TSTF-505 changes are incorporated.
One channel inoperable.	3.3.2.D.1	3.3.2.D.1	Yes	TSTF-505 changes are incorporated.
One Containment Pressure channel inoperable.	3.3.2.E.1	3.3.2.E.1.1	Yes	PINGP Condition E is “One or more Containment Pressure channel(s) inoperable”. This Condition is not in the TSTF-505 TS as the TSTF-505 exclusion criteria eliminate the equivalent NUREG-1431 TS 3.3.2, Condition E. The wording of the PINGP TS varies from that in NUREG-1431 (i.e., PINGP TS Required Action E.1.1 is to “place inoperable channel(s) in trip”, while the NUREG-1431 Required Action E.1 wording is “place channel in bypass”. NSPM proposes to apply a RICT to the existing PINGP TS 3.3.2, Required Action E.1.1, consistent with TSTF-505.
One channel or train inoperable.	3.3.2.F.1	3.3.2.F.1	Yes	TSTF-505 changes are incorporated.
One train inoperable.	3.3.2.G.1	3.3.2.G.1	Yes	TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
One train inoperable.	3.3.2.H.1	-	No	The PINGP TS do not contain this Condition. Therefore, TSTF-505 changes are not incorporated.
One channel inoperable.	3.3.2.I.1	3.3.2.H.1	Yes	TSTF-505 changes are incorporated.
[PINGP TS Condition] One or both channel(s) inoperable on one bus.	-	3.3.2.I.1	Yes	This is a PINGP-specific Condition. Both channels inoperable represents a loss of function, therefore, NSPM proposes adding a note to limit applicability of a RICT to one channel inoperable. Therefore, with the note, changes consistent with TSTF-505 are incorporated.
One Main Feedwater Pumps trip channel inoperable.	3.3.2.J.1	-	N/A	The PINGP TS do not contain this Condition. Therefore, TSTF-505 changes are not incorporated.
One train inoperable.	3.3.2.K.1	3.3.2.J.1	No	The wording of TSTF-505 varies from the PINGP TS (i.e., the PINGP TS 3.3.2.J.1 Completion Time is "Immediately"). Therefore, TSTF-505 changes are not incorporated.
One channel inoperable.	3.3.2.L.1	3.3.2.K.1	No	The wording of TSTF-505 varies from the PINGP TS (i.e., the PINGP TS 3.3.2.K.1 Completion Time is "Immediately"). Therefore, TSTF-505 changes are not incorporated.
Required Action and associated Completion Time of Conditions B or C, or K not met.	[NEW] 3.3.2.M	[NEW] 3.3.2.L	No	The PINGP TS do not currently contain this Condition. This new Condition will be added based on TSTF-505 and consistent with the existing PINGP TS.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Required Action and associated Completion Time of Conditions D, E, F, G, or L not met.	[NEW] 3.3.2.N	[NEW] 3.3.2.M	No	The PINGP TS do not currently contain this Condition. This new Condition will be added based on TSTF-505 and consistent with the existing PINGP TS.
Required Action and associated Completion Time of Condition H, I, or J not met.	[NEW] 3.3.2.O	[NEW] 3.3.2.N	No	The PINGP TS do not currently contain this Condition. This new Condition will be added based on TSTF-505 and consistent with the existing PINGP TS.
Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	**3.3.5**	**3.3.4**		PINGP TS is titled “4 kV Safeguards Bus Voltage Instrumentation”.
One or more Functions with one channel per bus inoperable.	3.3.5.A.1	3.3.4.A.1	No	The wording of PINGP TS Required Action differs from that in TSTF-505 and is excluded by the criteria in TSTF-505. Therefore, TSTF-505 changes are not incorporated.
One or more Functions with two or more channels per bus inoperable.	3.3.5.B.1	3.3.4.B.1 3.3.4.B.2	No	The wording of PINGP TS Required Actions differs from that in TSTF-505 and are excluded by the criteria in TSTF-505. Therefore, TSTF-505 changes are not incorporated.
[PINGP TS Condition Description] One required automatic load sequencer inoperable.	3.8.1.F.1	3.3.4.C.5	Yes	PINGP TS 3.3.4 Required Action C.5 is equivalent to TS 3.8.1 Required Action F.1 in TSTF-505. Therefore, TSTF-505 changes are incorporated.
Boron Dilution Protection System (BDPS)	**3.3.9**			
One train inoperable.	3.3.9.A.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Reactor Coolant System (RCS) Loops – Mode 3	**3.4.5**	**3.4.5**		
One required RCS loop inoperable.	3.4.5.A.1	3.4.5.A.1	No	The proposed PINGP RICT Program is applicable in Modes 1 and 2. This TS is applicable in Mode 3. Therefore, TSTF-505 changes are not incorporated.
One required RCS loop not in operation with Rod Control System capable of rod withdrawal.	3.4.5.C.1 3.4.5.C.2	3.4.5.C.1 3.4.5.C.2	No No	The proposed PINGP RICT Program is applicable in Modes 1 and 2. This TS is applicable in Mode 3. Therefore, TSTF-505 changes are not incorporated.
Pressurizer	**3.4.9**	**3.4.9**		
One group of pressurizer heaters inoperable.	3.4.9.B.1	3.4.9.B.1	Yes	TSTF-505 changes are incorporated.
Pressurizer Power Operated Relief Valves (PORVs)	**3.4.11**	**3.4.11**		
One [or two] PORV[s] inoperable and not capable of being manually cycled.	3.4.11.B.3	3.4.11.B.3	Yes	PINGP Condition B is “One PORV inoperable and not capable of being manually cycled”. TSTF-505 changes are incorporated.
One [or two] block valve(s) inoperable.	3.4.11.C.2	3.4.11.C.2	Yes	PINGP Condition C is “One block valve inoperable”. TSTF-505 changes are incorporated.
ECCS – Operating	**3.5.2**	**3.5.2**		
One or more trains inoperable.	3.5.2.A.1	3.5.2.A.1	Yes	TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Containment Air Locks	**3.6.2**	**3.6.2**		
One or more containment air locks inoperable for reasons other than Condition A or B.	3.6.2.C.3	3.6.2.C.3	Yes	TSTF-505 changes are incorporated.
Containment Isolation Valves	**3.6.3**	**3.6.3**		
One or more penetration flow paths with one containment isolation valve inoperable [for reasons other than Condition[s] D [and E]].	3.6.3.A.1	3.6.3.A.1	Yes	PINGP Condition A is "One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D". TSTF-505 changes are incorporated.
One or more penetration flow paths with one containment isolation valve inoperable.	3.6.3.C.1	3.6.3.C.1	Yes	TSTF-505 changes are incorporated.
Containment Spray and Cooling Systems	**3.6.6A**	**3.6.5**		
One containment spray train inoperable.	3.6.6A.A.1	3.6.5.A.1	Yes	TSTF-505 changes are incorporated.
One [required] containment cooling train inoperable.	3.6.6A.C.1	3.6.5.C.1	Yes	PINGP Condition C is "One or both containment cooling fan coil unit(s) (FCU) in one train inoperable". Changes consistent with TSTF-505 are incorporated.
Two [required] containment cooling trains inoperable.	3.6.6A.D.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.
[PINGP TS Condition Description] One containment cooling FCU in each train inoperable.		3.6.5.D.2	Yes	This is a PINGP-specific Condition to which NSPM proposes to apply a RICT. Changes consistent with TSTF-505 are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Hydrogen Ignition System (HIS) (Ice Condenser)	**3.6.10**	**-**		
One HIS train inoperable.	3.6.10.A.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.
One containment region with no OPERABLE hydrogen ignitor.	3.6.10.B.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.
Air Return System (ARS) (Ice Condenser)	**3.6.14**	**-**		
One ARS train inoperable.	3.6.14.A.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.
Ice Condenser Doors (Ice Condenser)	**3.6.16**	**-**		
One or more ice condenser inlet doors inoperable due to being physically restrained from opening.	3.6.14.A.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.
One or more ice condenser doors inoperable for reasons other than Condition A or not closed.	3.6.14.B.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.
Divider Barrier Integrity (Ice Condenser)	**3.6.17**	**-**		
One or more personnel access doors or equipment hatches open or inoperable, other than for personnel transit entry.	3.6.17.A.1	-	No	The PINGP TS do not contain this TS. Therefore, TSTF-505 changes are not incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Main Steam Safety Valves (MSSVs)	**-**	**3.7.1**		
One MSSV inoperable.	-	3.7.1.A.1	Yes	This Condition is not in the TSTF-505 TS as the TSTF-505 exclusion criteria eliminate the equivalent NUREG-1431 TS 3.7.1, Condition B. The wording of the PINGP TS varies from that in NUREG-1431 (i.e., PINGP TS Required Action is to “restore MSSV to OPERABLE status”, while the NUREG-1431 wording is based upon graduated power levels.
Main Steam Isolation Valves (MSIVs)	**3.7.2**	**3.7.2**		
One MSIV inoperable in MODE 1.	3.7.2.A.1	3.7.2.A.1	Yes	TSTF-505 changes are incorporated.
Atmospheric Dump Valves (ADVs)	**3.7.4**	**3.7.4**		PINGP TS is titled “Steam Generator (SG) Power Operated Relief Valves (PORVs)”.
One required ADV line inoperable.	3.7.4.A.1	3.7.4.A.1	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “SG PORV line” instead of “ADV line” in TSTF-505). TSTF-505 changes are incorporated.
Auxiliary Feedwater (AFW) System	**3.7.5**	**3.7.5**		
One steam supply to turbine driven AFW pump inoperable.	3.7.5.A.1	3.7.5.A.1	Yes	TSTF-505 changes are incorporated.
One AFW train inoperable in MODE 1, 2, or 3 [for reasons other than Condition A].	3.7.5.B.1	3.7.5.B.1	Yes	TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Component Cooling Water (CCW) System	**3.7.7**	**3.7.7**		PINGP TS is titled “Component Cooling Water (CC) System”.
One CCW train inoperable.	3.7.7.A.1	3.7.7.A.1	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “CC” instead of “CCW” in TSTF-505). TSTF-505 changes are incorporated.
Service Water System (SWS)	**3.7.8**	**3.7.8**		PINGP TS is titled “Cooling Water (CL) System”.
[PINGP TS Condition Description] No safeguards CL pumps OPERABLE for one train.	-	3.7.8.A.1	Yes	PINGP TS 3.7.8 Condition A is a PINGP-specific condition. NSPM proposes to apply a RICT to the existing PINGP TS 3.7.8, Required Action A.1, consistent with TSTF-505.
One SWS train inoperable.	3.7.8.A.1	3.7.8.B.3	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “CL supply header” instead of “SWS train” in TSTF-505). TSTF-505 changes are incorporated.
Ultimate Heat Sink (UHS)	**3.7.9**	**-**		PINGP TS is titled “Emergency Cooling Water (CL) Supply”.
One or more cooling towers with one cooling tower fan inoperable.	3.7.9.A.1	-	No	The PINGP TS do not contain this Condition. Therefore, TSTF-505 changes are not incorporated.
AC Sources – Operating	**3.8.1**	**3.8.1**		
One [required] offsite circuit inoperable.	3.8.1.A.3	3.8.1.A.2	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “path” instead of “offsite circuit” in TSTF-505). TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
One [required] DG inoperable.	3.8.1.B.4	3.8.1.B.4	Yes	TSTF-505 changes are incorporated.
Two [required] offsite circuits inoperable.	3.8.1.C.2	3.8.1.C.2	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “path” instead of “offsite circuit” in TSTF-505). TSTF-505 changes are incorporated.
One [required] offsite circuit inoperable. <u>AND</u> One [required] DG inoperable.	3.8.1.D.1 3.8.1.D.2	3.8.1.D.1 3.8.1.D.2	Yes Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “path” instead of “offsite circuit” in TSTF-505). TSTF-505 changes are incorporated.
One [required] [automatic load sequencer] inoperable.	3.8.1.F.1	3.3.4.C.5	Yes	See previous comment in TSTF-505 TS 3.3.5, “Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation”.
DC Sources – Operating	**3.8.4**	**3.8.4**		
One [or two] battery charger[s on one train] inoperable.	3.8.4.A.3	3.8.4.A.4	Yes	PINGP Condition A is “One battery charger inoperable”. TSTF-505 changes are incorporated.
One [or two] batter[y][ies on one train] inoperable.	3.8.4.B.1	3.8.4.B.4	Yes	PINGP Condition B is “One battery inoperable”. TSTF-505 changes are incorporated.
One DC electrical power subsystem inoperable for reasons other than Condition A [or B].	3.8.4.C.1	3.8.4.C.1	Yes	TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Inverters – Operating	**3.8.7**	**3.8.7**		
One [required] inverter inoperable.	3.8.7.A.1	3.8.7.A.1	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS uses “Reactor Protection Instrument AC inverter” instead of “inverter” in TSTF-505). TSTF-505 changes are incorporated.
Distribution Systems-Operating	**3.8.9**	**3.8.9**		
One or more AC electrical power distribution subsystems inoperable.	3.8.9.A.1	3.8.9.A.1	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS contains “safeguards” in front of “AC electrical power distribution subsystems” in TSTF-505). TSTF-505 changes are incorporated.
One or more DC electrical power distribution subsystems inoperable.	3.8.9.C.1	3.8.9.B.1	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS contains “safeguards” in front of “DC electrical power distribution subsystems” in TSTF-505). TSTF-505 changes are incorporated.
One or more AC vital buses inoperable.	3.8.9.B.1	3.8.9.C.1	Yes	Wording of PINGP TS differs from TSTF-505 (i.e., PINGP TS contains “Reactor Protection Instrument AC panel” instead of “AC vital buses” in TSTF-505). TSTF-505 changes are incorporated.

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

TSTF-505 Tech Spec Section Title / Condition Description	TSTF-505 TS	PINGP TS	Apply RICT?	Comments
Programs and Manuals	**5.5**	**5.5**		
Programs and Manuals	5.5.18	5.5.18	No	The PINGP TS do not currently contain this program. The new RICT Program will be added to the PINGP TS 5.5.18 consistent with TSTF-505.

ATTACHMENT 5

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

PINGP RICT PROGRAM PRA IMPLEMENTATION ITEMS

(1 Page Follows)

RICT Program PRA Implementation Items

1.0 INTRODUCTION

The table below identifies the items that are required to be completed prior to implementation of the Risk Informed Completion Time (RICT) Program at PINGP, Units 1 and 2. All issues identified below will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT Program.

Table A5-1: RICT Program PRA Implementation Items

No.	Implementation Items
1.	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 PINGP units prior to implementation of the RICT Program.
2.	NSPM shall ensure that the High-High Containment Pressure signal input to the MSIV closure logic is modeled in the PINGP PRA prior to implementation of the RICT Program.

ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

LIST OF REVISED REQUIRED ACTIONS TO CORRESPONDING PRA FUNCTIONS

(33 Pages Follow)

List of Revised Required Actions to Corresponding PRA Functions

1.0 INTRODUCTION

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

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

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

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

- Column “Tech Spec Description”: Lists all of the LCOs and condition statements within the scope of the RICT Program.
- Column “SSCs Covered by TS LCO Condition and Applicable Mode(s)”: The SSCs addressed by each action requirement and the Modes in which they apply relative to the PINGP RICT Program.
- Column “Modeled in PRA?”: Indicates whether the SSCs addressed by the TS LCO Condition are included in the PRA.
- Column “Function Covered by TS LCO Condition”: Lists a summary of the required functions from the design basis analyses.

- Column “Design Success Criteria”: A summary of the success criteria from the design basis analyses.

- Column “PRA Success Criteria”: The function success criteria modeled in the PRA.

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

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

Examples of calculated RICT are provided in Table E1-2 for each individual Condition to which the RICT applies (assuming no other SSCs modeled in the PRA are unavailable). These example calculations demonstrate the scope of the SSCs covered by TSs modeled in the PRA. RICTS were calculated for both units and while the results were generally similar, the most limiting RICT is shown in Table E1-2. Also note that the more limiting of the core damage frequency (CDF) and large early release frequency (LERF) RICT result is shown.

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

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.1.B	One Manual Reactor Trip channel inoperable.	Two Manual Reactor Trip channels (Mode 1 & 2)	Yes	Reactor Trip Initiation	One of two Manual Reactor Trip channels	Same	(Note 4)
3.3.1.D	One Power Range Neutron Flux channel inoperable.	Four Power Range Neutron Flux-High channels (Mode 1 & 2) Four Power Range Neutron Flux-Low channels (Mode 1, below P-10 & 2) Four Power Range Neutron Flux High Positive Rate channels (Mode 1 & 2) Four Power Range Neutron Flux High Negative Rate channels (Mode 1 & 2)	Yes	Reactor Trip Initiation	Two of four Power Range Neutron Flux-High channels Two of four Power Range Neutron Flux-Low channels Two of four Power Range Neutron Flux High Positive Rate channels Two of four High Negative Rate channels	Same	(Notes 1 and 2)
3.3.1.E	One channel inoperable.	Four Overtemperature ΔT channels (Mode 1 & 2) Four Overpower ΔT channels	Yes	Reactor Trip Initiation	Two of four Overtemperature ΔT channels Two of four Overpower ΔT	Same	(Notes 1 and 2)

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		(Mode 1 & 2) Three Pressurizer Pressure High channels (Mode 1 & 2) Three Steam Generator Water Level – Low Low channels per SG (Mode 1 & 2)			channels Two of three Pressurizer Pressure High channels Two of three Steam Generator Water Level Low-Low channels on either SG		
3.3.1.K	One channel inoperable.	Four Pressurizer Pressure Low channels (Mode 1, above P-7) Three Pressurizer Water Level – High channels (Mode 1, above P-7) Three Reactor Coolant Flow – Low channels per SG (Mode 1, above P-8)	Yes	Reactor Trip Initiation	Two of four Pressurizer Pressure Low channels Two of three Pressurizer Water Level – High channels Two of three Reactor Coolant Flow – Low channels on either RCS loop	Same	(Notes 1 and 2)
3.3.1.L	One or both channel(s) inoperable on one bus.	Two Under-frequency channels per 4 kV Bus (Buses 11/12 and 21/22) (Mode 1, above P-8) Two Under-voltage channels per 4 kV Bus	Yes	Reactor Trip Initiation	One of two Under-frequency channels on two of two buses One of two Under-voltage channels on two of two buses	Same for Under-voltage; Under-frequency is not directly modeled.	Under-voltage channels are modeled, are logically equivalent, and have the same component failure rate and

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		11 and 12 (21 and 22) (Mode 1, above P-7)					can be used as a surrogate.
3.3.1.M	One Reactor Coolant Pump Breaker Open channel inoperable.	One RCP Breaker Open channel per RCP Breaker (Mode 1, above P-7)	Yes	Reactor Trip Initiation	One of two RCP Breaker position channels (above P-8) Two of two RCP Breaker Position channels (between P-7 and P-8)	The PRA conservatively assumes that two of two logic is always applicable, regardless of power level.	
3.3.1.O	One Turbine Trip channel inoperable.	Three Low Autostop Oil Pressure channels (Mode 1, above P-9) Two Turbine Stop Valve Closure channels (Mode 1, above P-9)	Yes	Reactor Trip Initiation	Two of three Low Autostop Oil Pressure channels Two of two Turbine Stop Valve Closure channels	Same	
3.3.1.Q	One train inoperable.	Two Trains of Safety Injection (SI) Input from ESFAS (Mode 1 & 2) Two trains of RTS Automatic Trip Logic (Mode 1 & 2)	Yes	Reactor Trip Initiation	One of two trains of SI Input from ESFAS One of two trains of RTS Automatic Trip Logic	Same	
3.3.1.R	One RTB train inoperable.	Two trains of Reactor Trip Breakers and Bypass Breakers (Mode 1 & 2)	Yes	Reactor Trip Initiation	One of two RTB trains	Same	(Note 5)

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.1.V	One trip mechanism inoperable for one RTB.	One Reactor Trip Breaker Undervoltage Mechanism and One Shunt Trip Mechanism per RTB (Mode 1 & 2)	Yes	Reactor Trip Initiation	One trip mechanism	Same	(Note 4) Undervoltage and shunt trip are within the component boundary of the RTB, which is modeled.
3.3.2.B	One channel or train inoperable.	<u>SI Function:</u> Two SI Manual Initiation channels (Mode 1 & 2) <u>Containment Spray (CS) Function:</u> Two CS Manual Initiation channels (Mode 1 & 2) <u>Containment Isolation (CI) Function:</u> Two CI Manual Initiation channels (Mode 1 & 2)	Yes	ESF Actuation	<u>SI Function:</u> One of two SI Manual Initiation channels <u>CS Function:</u> Two of two CS Manual Initiation channels <u>CI Function:</u> One of two CI Manual Initiation channels	<u>SI Function:</u> Same <u>CS Function:</u> Not directly modeled. CS is screened out of the PRA. <u>CI Function:</u> Not directly modeled.	Manual SI can be used as a surrogate for Manual CI since SI signal generates CI signal. Hydraulic analysis has been performed which shows that CS success or failure does not impact which sequences contribute to LERF.
3.3.2.C	One train inoperable.	<u>SI Function:</u> Two SI Automatic Actuation Logic trains (Modes 1 & 2)	Yes	ESF Actuation	<u>SI Function:</u> One of two SI Automatic Actuation Logic trains	<u>SI Function:</u> Same	Hydraulic analysis has been performed which shows

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		CS Function: Two CS Automatic Actuation Logic trains (Modes 1 & 2) CI Function: Two CI Automatic Actuation Logic trains (Modes 1 & 2)			CS Function: One of two SI Automatic Actuation Logic trains CI Function: One of two CI Automatic Actuation Logic trains	CS Function: Not directly modeled. CS is screened out of the PRA. CI Function: Same	that CS success or failure does not impact which sequences contribute to LERF.
3.3.2.D	One channel inoperable.	SI Function: Three High Containment Pressure channels (Mode 1 & 2) Three Pressurizer Low Pressure channels (Mode 1 & 2) Three Steam Line Low Pressure channels per steam line (Mode 1 & 2) Steam Line Isolation (SLI) Function: Three High-High Containment Pressure channels (Mode 1; Mode 2, except when both Main Steam	Yes	ESF Actuation SLI AFW pump start	SI Function: Two of three High Containment Pressure channels Two of three Pressurizer Low Pressure channels Two of three Steam Line Low Pressure channels per steam line SLI Function: Two of three High-High Containment Pressure channels One of two High	Same	(Note 2) The High-High Containment Pressure channels are not currently modeled in PRA, but will be added prior to implementation of the RICT Program (see Attachment 5 of this LAR).

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		Isolation Valves (MSIVs) are closed) Two High Steam Flow channels per steam line, SI Function channels (see above), and four Low-Low Reactor Coolant System (RCS) T_{avg} channels (Mode 1; Mode 2, except when both MSIVs are closed Two High-High Steam Flow Channels per steam line and SI Function channels (see above) (Mode 1; Mode 2, except when both MSIVs are closed) <u>Auxiliary Feedwater (AFW) Function:</u> Three Low-Low SG Water Level channels per SG (Mode 1 & 2)			Steam Flow channels per steam line coincident with SI Function (see above) and coincident with two of four Low-Low RCS T_{avg} channels One of two High-High Steam Flow channels per steam line coincident with SI Function (see above) <u>AFW Function:</u> Two of three Low-Low SG Water Level channels on one of two SGs.		
3.3.2.E	One or more Containment Pressure	<u>CS Function:</u> Six (three sets of two) High-High Containment	Yes	CS	<u>CS Function:</u> One of two High-High Containment Pressure	<u>CS Function:</u> Not directly modeled. CS is	Hydraulic analysis has been performed

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	channel(s) inoperable.	Pressure channels (Mode 1 & 2)			channels in three of three sets	screened out of the PRA.	which shows that CS success or failure does not impact which sequences contribute to LERF.
3.3.2.F	One channel or train inoperable.	<u>SLI Function:</u> One Manual Initiation channel per loop (Mode 1; Mode 2, except when both MSIVs are closed)	Yes	SLI	<u>SLI Function:</u> One of one Manual Initiation channel per loop	Same	
3.3.2.G	One train inoperable.	<u>SLI Function:</u> Two trains of Automatic Actuation Relay Logic (Mode 1; Mode 2, except when both MSIVs are closed) <u>Feedwater Isolation Function:</u> Two trains of Automatic Actuation Relay Logic (Mode 1; Mode 2, except when all Main Feedwater Regulation Valves (MFRV) and MFRV bypass valves are closed and de-	Yes	SLI and Feedwater Isolation	<u>SLI Function:</u> One of two SI Automatic Actuation Logic trains <u>Feedwater Isolation Function:</u> One of two SI Automatic Actuation Logic trains	Same	

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		activated or isolated by closed manual valve)					
3.3.2.H	One channel inoperable.	Feedwater Isolation Function: Three High-High Steam Generator Water Level channels per SG (Mode 1; Mode 2, except when all Main Feedwater Regulation Valves (MFRV) and MFRV bypass valves are closed and de-activated or isolated by closed manual valve)	Yes	Feedwater Isolation	Feedwater Isolation Function: Two of three High-High SG Water Level channels per SG	Same	
3.3.2.I	One or both channel(s) inoperable on one bus.	AFW Function: Two Undervoltage channels per 4 kV Bus (Buses 11/12 and 21/22) (Mode 1 & 2)	Yes	AFW pump start	AFW Function: One of two Undervoltage channels on two of two buses	Same	
3.3.4.C	One required automatic load sequencer inoperable.	One Automatic Load Sequencer per 4 kV Bus (Buses 11/12 and 21/22) (Mode 1 & 2)	Yes	4 kV bus load shedding, sequencing, and Diesel Generator start	One load sequencer per bus	Same	
3.4.9.B	One group of pressurizer heaters	Two Groups of safeguards powered Pressurizer Heaters	Yes	RCS Subcooling Margin	One of two groups of safeguards powered pressurizer heaters,	One out of five groups (two safeguards	Modeled for long-term secondary

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	inoperable.	(Mode 1 & 2)			with a capacity of ≥100kW	powered; three non-safeguards powered) of pressurizer heaters under normal conditions. If offsite power is lost, the PRA success criterion is the same as design success criterion.	cooling success only.
3.4.11.B	One PORV inoperable and not capable of being manually cycled.	Two Pressurizer Power Operated Relief Valves (PORVs) (Mode 1 & 2)	Yes	RCS depressurization, feed and bleed	Two PORVs	One PORV	Manual PORV operation credited for feed and bleed cooling and cooldown and depressurization after a small loss of coolant accident (SLOCA) or a steam generator tube rupture (SGTR).

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.4.11.C	One block valve inoperable.	Two Pressurizer PORV block valves (Mode 1 & 2)	Yes	Isolate associated PORV	Two PORV block valves closable	Same	
3.5.2.A	One or more trains inoperable.	Two ECCS trains (SI and RHR in each train) (Mode 1 & 2)	Yes	Emergency RCS makeup via injection from the RWST to the cold legs and upper plenum, and recirculation from the containment sump to the upper plenum or the SI pump suction.	With One Train Inoperable: One of two SI pumps and one of two RHR pumps OPERABLE; One or More Pumps Inoperable: Two of two SI and/or two of two RHR pumps inoperable, but with a capability equivalent to ≥ 100% of a single OPERABLE ECCS train.	Same	
3.6.2.C	One or more containment air locks inoperable for reasons other than Condition A or B.	Containment Airlocks (Mode 1 & 2)	Yes	Containment Integrity	One of two containment air lock doors closed with acceptable containment leakage per LCO 3.6.1	Same	
3.6.3.A	One or more penetration flow paths with one	Two containment isolation valves per penetration	Yes	Containment boundary and minimization of	One of two isolation valves per penetration	Same, for modeled penetrations.	Only penetrations that can contribute to

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	containment isolation valve inoperable for reasons other than Condition D.	(Mode 1 & 2)		RCS inventory loss			LERF are modeled
3.6.3.C	One or more penetration flow paths with one containment isolation valve inoperable.	One containment isolation valve per penetration (Mode 1 & 2)	Yes	Containment boundary and minimization of RCS inventory loss	One of one isolation valve per penetration	Same	Only penetrations that can contribute to LERF are modeled
3.6.5.A	One containment spray train inoperable.	Two Containment Spray trains (Mode 1 & 2)	No	Containment cooling via injection from the RWST to the containment spray headers.	One of two containment spray trains	None	Hydraulic analysis has been performed to show that success or failure does not impact which sequences contribute to LERF.
3.6.5.C	One or both containment cooling fan coil unit(s) (FCU) in one train inoperable.	Two Containment Fan Coil trains (two FCUs per train) (Mode 1 & 2)	No	Containment cooling via heat transfer from the atmosphere to the cooling water system.	Two of four containment fan coil units	None	Hydraulic analysis has been performed to show that success or failure does not impact which

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
							sequences contribute to LERF.
3.6.5.D	One containment cooling FCU in each train inoperable.	Two Containment Fan Coil trains (two FCUs per train) (Mode 1 & 2)	No	Containment cooling via heat transfer from the atmosphere to the cooling water system.	Two of four containment fan coil units	None	Hydraulic analysis has been performed to show that success or failure does not impact which sequences contribute to LERF.
3.7.1.A	One MSSV inoperable.	Five Main Steam Safety Valves (MSSVs) (Mode 1 & 2)	Yes	Overpressure protection for the secondary system, SG overpressure protection, alternative heat sink for RCS overpressure protection	Five of five MSSVs per SG	One of five MSSVs per SG when associated PORV and steam dump not available	The SG PORVs, steam dump to condenser, and MSSVs are all credited in the PRA for steam relief to support secondary cooling.
3.7.2.A	One MSIV inoperable in MODE 1.	Two Main Steam Isolation Valves (MSIVs) (one MSIV per steam line) (Mode 1; Mode 2 except when both MSIVs are	Yes	Isolate Main Steam Lines	One MSIV closure per steam generator	Same	

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		closed)					
3.7.4.A	One SG PORV line inoperable.	Steam Generator Power Operated Relief Valves (PORV) (Mode 1 & 2)	Yes	Pressure relief and plant cooldown	One of two SG PORVs		Pressure Relief: The SG PORVs, steam dump to condenser, and MSSVs are all credited in the PRA for steam relief to support secondary cooling. Plant Cooldown: The SG PORVs and steam dump to condenser are credited in the PRA for plant cooldown.
3.7.5.A	One steam supply to turbine driven AFW pump inoperable.	Two steam supplies to the turbine driven AFW (TDAFW) pump (Mode 1 & 2)	Yes	Supply steam to support TDAFW pump operation	One of two steam flowpaths from the SGs to the TDAFW pump	Same	
3.7.5.B	One AFW train inoperable in MODE 1, 2, or 3 for reasons other than	Two AFW trains each comprised of one pump (one containing a motor driven AFW pump and the other a TDAFW	Yes	Supply feedwater to steam generators to remove RCS	One of two AFW trains (pumps or flow path) supplying feedwater to both SGs	Non-ATWS: One of two AFW pumps supplying feedwater to	

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	Condition A.	pump), piping, valves, and controls (Mode 1 & 2)		decay heat		one of two SGs. ATWS: One of two AFW pumps supplying feedwater to two of two SGs.	
3.7.7.A	One CC train inoperable.	Two CC trains each comprised of one pump with associated surge tank, piping, valves, heat exchanger, instrumentation, and controls (Mode 1 & 2)	Yes	Heat sink for removing process and operating heat from safety related components	One of two CC trains	Same	
3.7.8.A	No safeguards CL pumps OPERABLE for one train.	Two diesel-driven CL pumps (DDCLPs) and one motor-driven CL pump (121 MDCLP) (Mode 1 & 2)	Yes	Supply cooling water to the CL pump discharge header	One of two DDCLPs (or 121 MDCLP, if aligned).	Varies; see Section 2.4.7 of Attachment 1 of this LAR	
3.7.8.B	One CL supply header inoperable.	Two CL supply headers each consisting of piping, pumps, valves, instrumentation, and controls (Mode 1 & 2)	Yes	Supply cooling water to safety-related equipment and equipment for safe shutdown	One of two supply headers	Same	

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.8.1.A	One required path inoperable.	Two paths consisting of all breakers, transformers, switches, cabling, and controls to transmit power from the transmission network to the safeguards bus(es). (Mode 1 & 2)	Yes	Provide power from offsite transmission network to onsite safeguards buses	One qualified path to the grid for one safeguards bus	Same. When offsite power available.	(Note 7) The PRA model does not include components upstream of the 4kV bus source breaker. The loss of a path to the grid can be modeled by failing applicable source breaker(s) open.
3.8.1.B	One DG inoperable.	Two DGs capable of supplying onsite safeguards bus(es). (Mode 1 & 2)	Yes	Provide power to safeguards buses when offsite power to them is lost	One of two DGs	Same. When offsite power not available.	(Note 7) PRA success criteria also includes credit for re-powering buses through the cross-tie to the opposite unit in some circumstances.
3.8.1.C	Two paths inoperable.	Two paths consisting of all breakers, transformers, switches, cabling, and controls to transmit power from the transmission network to the safeguards bus(es)	Yes	Provide power from offsite transmission network to onsite safeguards buses	One qualified path to the grid for one safeguards bus	Same. When offsite power available.	(Note 7) The PRA model does not include components upstream of the 4kV bus source breaker. The

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		(Mode 1 & 2)					loss of a path to the grid can be modeled by locking applicable source breaker(s) open.
3.8.1.D	One path inoperable. <u>AND</u> One DG inoperable.	Two paths consisting of all breakers, transformers, switches, cabling, and controls to transmit power from the transmission network to the safeguards bus(es) and two DGs capable of supplying onsite safeguards bus(es). (Mode 1 & 2)	Yes	Provide power to safeguards buses when offsite power to them is lost	One qualified path to the grid and one DG for one safeguards bus	<u>Offsite Power Available:</u> One path to the grid. <u>Offsite Power Not Available:</u> One DG for one safeguards bus.	(Note 7)
3.8.4.A	One battery charger inoperable.	Two battery chargers (one per DC safeguards electrical power subsystem train) (Mode 1 & 2)	Yes	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	One battery charger for one of two DC trains	Same	
3.8.4.B	One battery inoperable.	Two DC batteries (one per DC safeguards electrical power subsystem train)	Yes	Ensure availability of required DC power to shut	Battery for one of two DC trains with capacity to carry expected shutdown	Battery for one of two DC trains with capacity to	

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
		(Mode 1 & 2)		down the reactor and maintain it in a safe condition	loads for a period of 1 hour.	carry SBO loads for a period of 2.75 hours	
3.8.4.C	One DC electrical power subsystem inoperable for reasons other than Condition A or B.	Two DC electrical power subsystems each consisting of one DC battery, one battery charger, cabling, and controls (Mode 1 & 2)	Yes	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	One of two DC trains	Same	
3.8.7.A	One Reactor Protection Instrument AC inverter inoperable.	Four inverters per unit (Mode 1 & 2)	Yes	Ensure availability of required DC power to shut down the reactor and maintain it in a safe condition	One of four reactor protection inverters	Same	
3.8.9.A	One or more safeguards AC electrical power distribution subsystems inoperable.	Two safeguards AC electrical power distribution subsystems with buses and MCCs energized to proper voltage (Mode 1 & 2)	Yes	Provide AC power for vital buses	At least one of two AC power subsystems available	Same	
3.8.9.B	One or more safeguards DC electrical power distribution	Two safeguards DC electrical power distribution subsystems with panels energized to	Yes	Provide DC power for vital panels	At least one of two DC power subsystems available	Same	

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
	subsystems inoperable.	proper voltage (Mode 1 & 2)					
3.8.9.C	One Reactor Protection Instrument AC panel inoperable.	Four Reactor Protection Instrument AC power distribution panels energized to proper voltage (Mode 1 & 2)	Yes	Provide regulated AC power for instrument panels	At least one of four Reactor Protection Instrument panels available	Same	

Table E1-1 Notes:

1. The reactor protection system is segmented into four distinct but interconnected modules: field transmitters and process sensors, instrumentation current loops, reactor protection bistables, and reactor trip relays. Field transmitters provide measurements of the unit parameters to the Reactor Protection System via separate, redundant channels. The reactor protection bistables determine when applicable sensor setpoints are reached. The reactor trip relays are actuated by the bistables and determine whether the applicable 2/4 or 2/3 logic is satisfied to generate a reactor trip. The reactor trip signal consists of two redundant trains, to initiate a reactor trip or actuate Engineering Safety Functions.

2. Depending on the measured parameter, three or four instrumentation channels are provided to ensure protective action when required and to prevent inadvertent isolation resulting from instrumentation malfunctions. The output trip signal of each instrumentation channel initiates a trip logic. Failure of any one trip logic does not result in an inadvertent trip. Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If a parameter is used for input to the reactor protection system and a control function, four channels with a two-out-of-four logic are sufficient.

3. Each instrumentation channel provides input to both trains of the reactor protection system, which initiates a reactor trip on one-out-of-two logic. Each reactor protection system train provides input to the Reactor Trip Breakers (RTBs) by de-energizing the RTB undervoltage coils, which trips open the RTBs, tripping the reactor. One-out-of-two open RTBs will trip the reactor.

4. Each RTB is equipped with a shunt trip device that is energized to trip the RTB open upon receipt of a manual reactor trip signal, thus providing a redundant and diverse trip mechanism. Two Manual Reactor Trip channels provide the signal from reactor trip

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

PINGP TS	PINGP TS Description	SSCs Covered by TS LCO Condition and Applicable Mode(s)	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments

switches located in the Main Control Room to the RTBs.

5. A trip breaker train consists of all trip breakers associated with a single Reactor Trip System logic train that are racked in, closed, and capable of supplying power to the Rod Control System.

6. PRA Success Criteria for bleed and feed cooling requires 1 SI pump and 1 PORV. Each PORV requires power from its respective DC power subsystem to perform its safety function for feed and bleed.

7. The safeguards 4 kV Buses 15 and 16 serve engineered safety feature auxiliaries on Unit 1, and Buses 25 and 26 serve similar functions on Unit 2. The electrical loading of the safeguards 4 kV buses at PINGP is asymmetric, primarily for the loading of the 12 and 21 AFW pumps and the 121 MDCLP. The 12 AFW pump is powered from Bus 16 and the 21 AFW pump is powered from Bus 25. In addition, the 121 MDCLP is powered by Unit 2 safeguards Bus 27, which is supplied by either Unit 2 4 kV safeguards Bus 25 or Bus 26. The PINGP 4 kV safeguards buses have been analyzed which confirmed that the Unit 1 and 2 safeguards DGs are adequately sized to supply safe shutdown loads with one unit in LOOP conditions and the other in SBO conditions.

RICTs were calculated for both units and both trains when applicable and the most limiting RICT is specified in the Table E1-2. Results were generally similar between Unit 1 and Unit 2. Following implementation of the RICT Program, the actual RICT values will be calculated on a unit-specific basis, using the actual plant configuration and the current revision of the PRA model representing the as-built, as-operated condition of the plant, as required by NEI 06-09-A, Revision 0 and the NRC Final Safety Evaluation.

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

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

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

Tech Spec	LCO Condition	RICT Estimate
3.3.1.B	One Manual Reactor Trip channel inoperable.	30 Days
3.3.1.D	One Power Range Neutron Flux channel inoperable.	30 Days
3.3.1.E	One channel inoperable.	30 Days
3.3.1.K	One channel inoperable.	30 Days
3.3.1.L	One or both channel(s) inoperable on one bus.	30 Days
3.3.1.M	One Reactor Coolant Pump Breaker Open channel inoperable.	21 Days
3.3.1.O	One Turbine Trip channel inoperable.	30 Days
3.3.1.Q	One train inoperable.	30 Days
3.3.1.R	One RTB train inoperable.	30 Days
3.3.1.V	One trip mechanism inoperable for one RTB.	30 Days
3.3.2.B	One channel or train inoperable.	30 Days
3.3.2.C	One train inoperable.	30 Days
3.3.2.D	One channel inoperable.	30 Days
3.3.2.E	One or more Containment Pressure channel(s) inoperable.	30 Days(1)
3.3.2.F	One channel or train inoperable.	30 Days
3.3.2.G	One train inoperable.	30 Days
3.3.2.H	One channel inoperable.	30 Days
3.3.2.I	One or both channel(s) inoperable on one bus.	30 Days
3.3.4.C	One required automatic load sequencer inoperable.	11 Days
3.4.9.B	One group of pressurizer heaters inoperable.	20 Days

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

Tech Spec	LCO Condition	RICT Estimate
3.4.11.B	One PORV inoperable and not capable of being manually cycled.	24 Days
3.4.11.C	One block valve inoperable.	30 Days
3.5.2.A	One or more trains inoperable.	14 Days
3.6.2.C	One or more containment air locks inoperable for reasons other than Condition A or B.	30 Days
3.6.3.A	One or more penetration flow paths with one containment isolation valve inoperable for reasons other than Condition D.	30 Days
3.6.3.C	One or more penetration flow paths with one containment isolation valve inoperable.	30 Days
3.6.5.A	One containment spray train inoperable.	30 Days(1)
3.6.5.C	One or both containment cooling fan coil unit(s) (FCU) in one train inoperable.	30 Days(1)
3.6.5.D	One containment cooling FCU in each train inoperable.	30 Days(1)
3.7.1.A	One MSSV inoperable.	30 Days
3.7.2.A	One MSIV inoperable in MODE 1.	30 Days
3.7.4.A	One SG PORV line inoperable.	30 Days
3.7.5.A	One steam supply to turbine driven AFW pump inoperable.	11 Days
3.7.5.B	One AFW train inoperable in MODE 1, 2, or 3 for reasons other than Condition A.	11 Days
3.7.7.A	One CC train inoperable.	6 Days
3.7.8.A	No safeguards CL pumps OPERABLE for one train.	28 Days
3.7.8.B	One CL supply header inoperable.	6 Days
3.8.1.A	One required path inoperable.	30 Days
3.8.1.B	One DG inoperable.	30 Days
3.8.1.C	Two paths inoperable.	5 Days
3.8.1.D	One path inoperable. <u>AND</u> One DG inoperable.	9 Days
3.8.4.A	One battery charger inoperable.	26 Days
3.8.4.B	One battery inoperable.	No Entry(2)
3.8.4.C	One DC electrical power subsystem inoperable for reasons other than Condition A or B.	No Entry(2)
3.8.7.A	One Reactor Protection Instrument AC inverter inoperable.	30 Days

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

Tech Spec	LCO Condition	RICT Estimate
3.8.9.A	One or more safeguards AC electrical power distribution subsystems inoperable.	No Entry(2)
3.8.9.B	One or more safeguards DC electrical power distribution subsystems inoperable.	No Entry(2)
3.8.9.C	One Reactor Protection Instrument AC panel inoperable.	30 Days

Table E1-2 Notes:

1. Performance of a hydraulic analysis has shown that success or failure of the Containment Spray and/or FCUs does not impact which sequences contributed to LERF. Therefore, there is no risk impact to removing them from service.

2. 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.

2.0 ADDITIONAL JUSTIFICATION FOR SPECIFIC ACTIONS

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

NSPM's additional justification for each of the identified PINGP TS is provided below:

2.1 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.

Condition D: One Power Range Neutron Flux channel inoperable.

As indicated in Table E1-1, the Power Range Neutron Flux channels are explicitly modeled in the PINGP PRA. The PRA Success Criterion is two of four channels.

As described in Section 7.4.1.3.9.1, "Nuclear Flux", of the PINGP USAR:

> Four power range nuclear flux channels are provided for overpower protection. Isolated outputs from all four channels are averaged for automatic rod control. If any channel fails in such a way as to produce a low output, that channel would be incapable of proper overpower protection. In principle, the same failure may cause rod withdrawal and hence, overpower. Two out of four overpower trip logic ensures that even with a failed channel a two out of three logic remains available to provide an overpower trip.

In addition, the control system responds only to rapid changes in indicated nuclear flux; slow changes or drifts are compensated by the temperature control signals. Finally, an overpower signal from any nuclear channel blocks automatic rod withdrawal. The allowable value for this rod withdrawal stop is below the reactor trip allowable value.

These alarms and actions signify periodic monitoring of spatial power distribution or reduced power. Therefore, TS 3.3.1 Condition D meets the requirements for inclusion in the RICT Program.

2.2 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.
Revised Condition R:	One [Reactor Trip Breaker (RTB)] train inoperable.

As indicated in Table E1-1, the RTB trains are explicitly modeled in the PINGP PRA. The PRA Success Criteria is the same as the Design Success Criteria which is one of two RTB trains.

An RTB train consists of all trip breakers associated with a single RTS logic train that are racked in and capable of supplying power to the Rod Control System. Therefore, the train may consist of the main breaker or main breaker and bypass breaker, depending upon the system configuration. The RTBs and Automatic Trip Logic ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Two devices in each breaker receive signals from the reactor protection system, either of which will trip the RTBs; an undervoltage trip device and a shunt trip device. This design provides a passive trip device which will trip the reactor on loss of breaker control power or the receipt of a trip signal and a positive acting device which provides a backup if the passive device fails to trip the reactor trip breakers upon receipt of a trip signal.

TSTF-411, “Surveillance Test Interval Extensions for Components of the Reactor Protection System (WCAP-15376-P)” (Reference 4), has not been adopted at PINGP. However, as demonstrated in Enclosure 2 to this LAR, the PINGP internal events PRA model (including internal flooding) meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2, to fully support the requirements of the RICT Program.

In addition, as described in Section 7.11, “ATWS Mitigating System Actuation Circuitry/Diverse Scram System [AMSAC/DSS]”, of the PINGP USAR, the AMSAC/DSS system installed at PINGP Units 1 and 2 further mitigates the effects of a failure of reactor protection to trip the reactor in the event of an anticipated transient. The AMSAC/DSS performs a reactor protection function, but is not part of the RTS. This system trips the turbine and starts AFW in addition to inserting the Control Rods in response to ATWS conditions. The specific PINGP AMSAC/DSS design was reviewed and found acceptable by the NRC (Reference 5).

Therefore, TS 3.3.1 revised Condition R meets the requirements for inclusion in the RICT Program.

2.3 TS 3.4.9 – Pressurizer

LCO: The pressurizer shall be OPERABLE with…two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 100 kW and capable of being powered from an emergency power supply.
Condition B: One group of pressurizer heaters inoperable.

As indicated in Table E1-1, the Pressurizer Heaters are explicitly modeled in the PINGP PRA. The Design Success Criteria is for one of two groups of safeguards powered pressurizer heaters with a capacity of ≥ 100kW, for long-term secondary cooling success only. The PRA Success Criteria is for one out of five groups of pressurizer heaters under normal conditions. If offsite power is lost, the PRA success criterion is the same as design success criterion.

The function of the pressurizer heaters is to maintain the water in the pressurizer at saturation temperature and maintain a constant operating pressure. The capability to maintain and control system pressure is required to maintain subcooled conditions in the RCS and ensure the capability to remove core decay heat by either forced or natural circulation of reactor coolant.

The pressurizer heaters are not credited for pressure control in the analysis supporting the PRA success criteria. However, for conservatism the pressurizer heaters are explicitly modeled in the PRA to maintain pressure over the long-term in scenarios where the RCS remains intact and secondary cooling is successful. These scenarios represent the conditions where loss of heat from the pressurizer through conduction or loss of volume through RCP seal normal leakoff could eventually result in loss of subcooling margin in the RCS. Other scenarios where the RCS does not remain intact (e.g. LOCAs) or when secondary cooling is not successful and core cooling is maintained via primary feed and bleed do not credit pressurizer heaters because operation of the ECCS pumps will effectively maintain RCS pressure and subcooling when necessary to support secondary cooling.

Therefore, TS 3.4.9 Condition B meets the requirements for inclusion in the RICT Program.

2.4 TS 3.5.2 – ECCS – Operating

LCO: Two ECCS trains shall be OPERABLE.
Condition A: One or more trains inoperable.

As indicated in Table E1-1, the ECCS trains are explicitly modeled in the PINGP PRA. The PRA Success Criterion is the same as the Design Success Criteria which is one of two Safety Injection (SI) and one of two Residual Heat Removal (RHR) pumps.

Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS.

Additionally, PINGP TS 3.5.2 Condition C requires immediate entry into LCO 3.0.3 for “less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available”. Therefore, Condition C prevents an ECCS loss of function from occurring due to two ECCS trains being inoperable in Condition A. The PRA Success Criterion for LCO 3.5.2 also considers the condition where both SI and/or both RHR pumps are inoperable, but TS 3.5.2 Condition C is met. In this case, the Success Criterion for the PRA is modified to two of two SI and two of two RHR pumps, as applicable.

Therefore, TS 3.5.2 Condition A meets the requirements for inclusion in the RICT Program.

2.5 TS 3.6.2 – Containment Air Locks

LCO:	Two containment air locks shall be OPERABLE.
Condition C:	One or more containment air locks inoperable for reasons other than Condition A or B.

As indicated in Table E1-1, the containment air locks are modeled in the PINGP PRA. The PRA Success Criteria is the same as the Design Success Criteria which is one of two containment air lock doors closed with acceptable containment leakage per LCO 3.6.1. Failure of the containment airlock function is modeled as early containment bypass in the PRA.

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a design basis accident (DBA). The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident, as described in the PINGP USAR, Section 14. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Compliance with the remaining portions of TS 3.6.2 ensures 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 TS 3.6.2 requires the condition to be assessed in accordance with TS 3.6.1, “Containment” (i.e., “initiate action to evaluate overall containment leakage rate per LCO 3.6.1” with a Completion Time of immediately).

Therefore, TS 3.6.2 Condition C meets the requirements for inclusion in the RICT Program.

2.6 TS 3.6.5 – Containment Spray and Cooling Systems

LCO:	Two containment spray trains and two containment cooling trains shall be OPERABLE.
Condition A:	One containment spray train inoperable.
Condition C:	One or both containment cooling Fan Coil Unit(s) (FCU) in one train inoperable.
Condition D:	One containment cooling FCU in each train inoperable.

The function of the Containment Spray (CS) system is to provide a spray of cold borated water mixed with sodium hydroxide into the upper regions of containment to reduce the containment pressure and temperature and to remove fission products from the containment atmosphere during a DBA. The function of the containment FCUs is to cool the containment atmosphere to limit post-accident pressure and temperature to less than the design values.

The CS and FCU systems were evaluated in a PRA calculation which concluded that success or failure of the systems will not change existing non-Large Early Release (LER) sequences into LER sequences. Thus the success or failure of CS or containment FCUs to provide containment cooling has been screened out of the PRA and is not directly modeled. Adverse impacts caused by operation of the CS system are considered; such as increased Refueling Water Storage Tank (RWST) depletion rate during ECCS injection, potential for spurious operation and subsequent loss of RWST inventory after a fire initiating event, and potential failure of 4 kV bus load-rejection sequence if the CS breaker fails to open on demand.

Since the system success or failure does not impact which core damage sequences are classified as contributing to LERF, the quantified RICT will be based on the increase in CDF/LERF due to the seismic penalty factor and configuration-specific risk for the other unrelated equipment out of service during the period of time the RICT is active.

Therefore, TS 3.6.5 Conditions A, C, and D meet the requirements for inclusion in the RICT Program.

2.7 TS 3.7.2 – Main Steam Isolation Valves (MSIVs)

LCO:	Two MSIVs shall be OPERABLE.
Condition A:	One MSIV inoperable in MODE 1

The function of the MSIVs is to isolate steam flow from the secondary side of the steam generators following a main steam line break (MSLB). MSIV closure terminates flow from the unaffected (intact) SGs. Closing the MSIVs isolates each SG from the other, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the SGs.

As described in the PINGP USAR, Section 14.5.5.2, “Expected Plant Response, depending on the location of a main steam line break, isolation of the non-faulted SG is accomplished via the following:

- Breaks between the SG and MSIV: Successful closure of the non-faulted SG MSIV or successful closure of the faulted SG non-return check valve (NRCV) will isolate the non-faulted steam generator.

- Breaks between the MSIV and NRCV: Successful closure of the non-faulted SG MSIV or successful closure of the faulted SG NRCV will isolate the non-faulted SG.

- Breaks downstream of the MSIV: Successful closure of either SG MSIV will prevent both SGs from being faulted.

Therefore, the design of the MSIV/NRCV combination precludes the blowdown of more than one SG, assuming a single active component failure (e.g., the failure of one MSIV or NRCV to close).

Both the MSIVs (including automatic and manual closure logic) and NRCVs are modeled in the PRA.

Therefore, TS 3.7.2 Condition A meets the requirements for inclusion in the RICT Program.

3.0 EVALUATION OF INSTRUMENTATION AND CONTROL SYSTEMS

The following Instrumentation Technical Specifications (TS) Sections are included in the TSTF-505 application for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

The PINGP Technical Specifications (TS) 3.3, “INSTRUMENTATION”, LCOs were developed to assure that the PINGP facility maintains necessary redundancy and diversity. The reactor protection systems are designed in accordance with IEEE 279-1968. Furthermore, it is shown that the intent of the applicable criteria and codes at the time of construction, such as the GDCs referenced in Sections 1.2 and 1.5 of the PINGP Updated Safety Analysis Report and IEEE 279-1971 (Reference 6), recognized by regulatory agencies (principally the NRC) concerned with the safe generation of nuclear power are reasonably met and that there is reasonable assurance that these systems will facilitate the production of power in a manner that insures no undue risk to the health and safety of the public. The Engineered Safety Features Actuation System meets the single failure criterion as defined in IEEE Standard 279-1971.

TSTF-505 (Reference 4) sets forth the following as guidance for what is to be included in this enclosure:

> The description of proposed changes to the protective instrumentation and control features in TS Section 3.3, "Instrumentation," should confirm that at least one redundant or diverse means (other automatic features or manual action) to accomplish the safety functions (for example, reactor trip, SI, containment isolation, etc.) remains available during use of the RICT, consistent with the defense-in-depth philosophy as specified in RG 1.174. (Note that for each

application, the staff may selectively audit the licensing basis of the most risk-significant functions with proposed RICTs to verify that such diverse means exist.)

The following sections provide the justification that defense-in-depth is maintained for the applicable functions throughout the application of the RICT Program.

3.1 Reactor Trip System (RTS) Instrumentation

The RTS design creates defense-in-depth through the degree of redundancy for each of its channels for each Functional Unit.

- Each Functional Unit has multiple channels.
- Each Functional Unit will cause a reactor trip with 1/2, 2/3 or 2/4 tripped signals.
- A bypassed channel does not trip. It reduces the number of total available channels by 1, from 2/4 to 2/3, or from 2/3 to 2/2.
- When applicable, if 1 channel in the Functional Unit is out of service, then that channel may be placed in a tripped state, for example reducing the redundancy from 2/4 required tripped channels to 1/3 required tripped channels.

The RTS also employs diversity in the number and variety of different inputs which will initiate a reactor trip. A given reactor trip will typically be accompanied by several diverse reactor trip inputs from the RTS.

- Manual Reactor Trip – 1/2
- High Neutron Flux (Low Setpoint) – 2/4
- Power Range High Neutron Flux (High Setpoint) – 2/4
- Overtemperature ΔT – 2/4
- Overpower ΔT – 2/4
- Pressurizer Low Pressure – 2/4
- Pressurizer High Pressure – 2/3
- Pressurizer High Water Level – 2/3
- Reactor Coolant Low Flow – 2/3 per Loop
- Monitored Electrical Supply for Reactor Coolant Pumps:
    - RCP Bus Undervoltage / Underfrequency – 2/2 Buses Sensed by 1/2 Sensors per bus
- Safety Injection Signal
    - Manual – 1/2
    - Low Pressurizer Pressure – 2/3
    - Low Steam Pressure from Either Loop – 2/3
    - High Containment Pressure – 2/3
- Turbine – Generator Trip
    - Low Auto Stop Oil Pressure – 2/3
    - Stop Valve Closure Indication – 2/2
- Low–Low Steam Generator Water Level – 2/3, either Loop

- Intermediate Range Nuclear Flux – 1/2
- Source Range Nuclear Flux – 1/2
- Power Range High Positive Neutron Flux Rate – 2/4
- Power Range High Negative Neutron Flux Rate – 2/4

3.2 Engineered Safety Features Actuation System (ESFAS)

The ESFAS design creates defense-in-depth due to the redundancy of the channels for each Function.

- Each Function has multiple channels.
- Each Function will cause an actuation with 1/2, 2/3 or 2/4 tripped signals.
- A bypassed channel does not trip. It reduces the number of total available channels by 1, from 2/4 to 2/3.
- When applicable, if 1 channel in the Function is out of service, then the 1 channel can be placed in trip, reducing the redundancy from 2/4 to 1/3.

ESFAS also employs diversity in the number and variety of different inputs which will actuate the associated equipment.

- Containment Isolation Actuation
    - Safety Injection Signal
        - Manual – 1/2
        - Low Pressurizer Pressure – 2/3
        - Low Steam Pressure from Either Loop – 2/3
        - High Containment Pressure – 2/3
    - Manual Containment Isolation – 1/2
    - Containment Ventilation Isolation Actuation
        - High Activity Signal from Air Particulate Detector or Radiogas Detector
        - Manual
            - Containment Isolation – 1/2
            - Containment Spray Actuation – 2/2
            - Safety Injection – 2/2
- Engineered Safety Features Actuation
    - Safety Injection Signal
        - Manual – 1/2
        - Low Pressurizer Pressure – 2/3
        - Low Steam Pressure from Either Loop – 2/3
        - High Containment Pressure – 2/3
    - Containment Spray Signal
        - Three 1/2 (High-High) Containment Pressure Containment Spray in Coincidence
        - Manual Spray – 2/2
    - Containment Air Cooling Signal – Safety injection signal initiates starting of all fans, transfers containment fan coils from chilled water to cooling water and

closes the control rod drive mechanism coil supply and return valves in accordance with the Safety Injection Starting Sequence.
  - Steam Flow
    - Coincidence of high-high steam flow (1/2) in the respective line and safety injection signal
    - Coincidence of (1/2) high steam flow in the respective line and safety injection signal and (2/4) low–low T_{avg}
  - High Containment pressure, Main Steam Isolation Set Point – 2/3 Hi containment pressure main steam isolation signal
  - Manual, per steam loop – 1/1 per steam line
- Auxiliary Feedwater Actuation
  - Turbine driven pump
    - Low–Low level in either steam generator
    - Loss of voltage on 2/2 4KV buses or a trip of 2/2 main feedwater pumps
    - Safety Injection or ATWS Mitigating System Actuation Circuitry (AMSAC) System Actuation.
  - Motor Driven Pump
    - Low–Low level in either steam generator
    - Trip of 2/2 main feedwater pumps
    - Safety Injection Signal as Modified by a Load Rejection/Restoration sequence or AMSAC system actuation
- Main Feedwater Isolation
  - Close main feedwater control valves
    - Safety Injection Signal
    - Reactor trip coincident with Low T_{avg}
    - 2/3 High-High Steam Generator level closes the valves to the effected Steam Generator.
  - Close bypass feedwater control valves and trip main feedwater pumps
    - Safety Injection Signal
    - 2/3 High-High Steam Generator Level

4.0 REFERENCES

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

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

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

4. TSTF-411, “Surveillance Test Interval Extensions for Components of the Reactor Protection System (WCAP-15376-P)”, Revision 1, dated August 7, 2002 (ADAMS Accession No. ML022470164)

5. Letter from the NRC to NSP, “Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: Modification to ATWS Mitigating System Actuating Circuitry (TAC Nos. MA1675 and MA1676)”, dated September 22, 1998 (ADAMS Accession No. ML022260827)

6. Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations”, dated June 3, 1971

ENCLOSURE 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

INFORMATION SUPPORTING CONSISTENCY WITH REGULATORY GUIDE 1.200, REVISION 2

(9 Pages Follow)

**Information Supporting Consistency with
Regulatory Guide 1.200, Revision 2**

1.0 INTRODUCTION

The purpose of this enclosure is to provide information on the technical adequacy of the Prairie Island Nuclear Generating Plant (PINGP) Probabilistic Risk Assessment (PRA) internal events model (including internal flooding) and the PINGP fire PRA model in support of the license amendment request (LAR) to adopt TSTF-505, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b”, Revision 2 (Reference 1). The PINGP internal events (including internal flooding) and fire PRA models described within this LAR are the same as those described within Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), submittals regarding adoption of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (Reference 2), and modification of the list of required NFPA 805 modifications (Reference 3), respectively.

The current internal events model of record (including internal flooding) is a combined PRA model that represents both units. The PRA model is built with a common one-top fault tree, including individual basic events for both Unit 1 and Unit 2 components. The internal flooding PRA is integrated into the internal events model.

The fire PRA model of record is built using the internal events PRA as a base, with fire PRA specific fault tree modifications and additions such as spurious operation and alternate shutdown.

Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, Revision 0 (Reference 4), as clarified by the NRC final safety evaluation of this report (Reference 5), defines the technical attributes of a PRA model and its associated Configuration Risk Management Program (CRMP) tool required to implement this risk-informed application. Meeting these requirements satisfies Regulatory Guide (RG) 1.174, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2 (Reference 6), requirements for risk-informed plant-specific changes to a plant's licensing basis.

NSPM employs a multi-faceted approach to establishing and maintaining the technical adequacy and fidelity of PRA models for its nuclear generation sites. This approach includes both a PRA maintenance and update process procedure and the use of self-assessments and independent peer reviews.

Section 2.0 of this enclosure describes the overall approach used to perform the peer review findings closure reviews for the PINGP PRAs. Section 3.0 discusses the requirements related to the scope of the PINGP PRA internal events model (including internal flooding). Section 4.0 addresses the technical adequacy of the PINGP PRA full power internal events model including internal flood for this application. Section 5.0 addresses the technical adequacy of the PINGP Fire PRA model for this application.

2.0 PEER REVIEW FINDINGS CLOSURE PROCESS

All the PRA models described below have been peer reviewed to the requirements of RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2 (Reference 7), the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) RA-Sa-2009 PRA Standard (hereafter “ASME/ANS PRA Standard”), “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Reference 8), NEI 05-04, “Process for Performing PRA Peer Reviews Using the ASME PRA Standard (Internal Events)”, Revision 2 (Reference 9), and NEI 07-12, “Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines”, Revision 1 (Reference 8). The review and closure of all but one of the finding-level Facts and Observations (F&Os) (FSS-B2-01) from the peer reviews have been independently evaluated to confirm that the associated model changes did not constitute a model upgrade. These reviews included F&Os that were associated with “met” supporting requirements, as well as all F&Os associated with supporting requirements (SRs) that were met at the Capability Category (CC) I level. A focused-scope peer review was performed on the FPRA subsequent to the independent F&O closure reviews to address the open F&O that was assessed as an upgrade. Based on the peer review, this F&O is considered closed and therefore it is not needed to be addressed in risk-informed applications (Reference 26). Expectations regarding preparation for the review (NEI 05-04, Section 4.2) and conduct of the self-assessment by the host utility (NEI 05-04, Section 4.3) were addressed prior to conduct of these reviews. This included documentation by NSPM of resolution of the prior PRA peer review finding-level F&Os and preparation of the information required for this independent assessment. The documented bases for F&O closure provided by NSPM included a written assessment whether the resolution constituted PRA maintenance or PRA upgrade.

The multi-disciplinary teams of reviewers for each closure review met the independence and relevant peer reviewer qualifications requirements in the ASME/ANS PRA Standard and related guidance. A total of 31 internal events F&Os and 41 fire F&Os were assessed, each of which was assigned to at least two of the reviewers.

References 11, 12, and 13 provide additional details of the F&O closure reviews, including the approach taken:

- The process guidance in NEI 05-04, Section 4.6, was applicable to this review.
- The independent technical review team reviewed the documented bases for closure of the finding-level F&Os prepared by NSPM.
- The independent technical review team determined whether the finding-level F&Os in question had been adequately addressed and could be closed out by consensus.
- As part of this process each F&O was reviewed regarding whether the closure response represented PRA maintenance or a PRA upgrade.

- Section 3 of each F&O closure report specifically states that the closure review team concluded that all SRs where the F&Os have been closed are now “met” at CC II.
- Details of the F&O Closure review assessments are documented in Appendix A of the F&O Closure Reports. The assessment for each F&O includes the determination that each closed finding meets CC II for all the applicable SRs of the ASME/ANS PRA Standard, as endorsed by RG 1.200 Revision 2.

3.0 REQUIREMENTS RELATED TO SCOPE OF PINGP INTERNAL EVENTS (INCLUDING INTERNAL FLOODING) AND FIRE PRA MODELS

Both the internal events PRA model of record (MOR) and the internal fire PRA MOR are at-power models (i.e., they directly address plant configurations during plant Modes 1 and 2 of reactor operation). The models include both core damage frequency (CDF) and large early release frequency (LERF). Internal flooding is included in both the CDF and LERF internal events PRA models. As described previously, the internal events (including internal flooding) PRA model described within this LAR is the same as the one described within the NSPM submittal of the LAR to adopt 10 CFR 50.69.

4.0 SCOPE AND TECHNICAL ADEQUACY OF PINGP INTERNAL EVENTS AND INTERNAL FLOODING PRA MODEL

NEI 06-09-A requires that the PRA be reviewed to the guidance of RG 1.200 for a PRA which meets CC II for the supporting requirements of the internal events at power ASME/ANS PRA Standard. It also requires that deviations from these CCs relative to the Risk Informed Completion Time (RICT) Program be justified and documented.

The information provided in this section demonstrates that the PINGP internal events PRA model (including internal flooding) meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2.

The PINGP PRA was peer reviewed in November 2010 applying NEI 05-04, the ASME/ANS PRA Standard and RG 1.200, Revision 2. The purpose of this review was to provide a method for establishing the technical adequacy of the PRA for the spectrum of potential risk-informed plant licensing applications for which the PRA may be used. The 2010 PINGP PRA peer review (Reference 14) was a full-scope review of the technical elements of the internal events, at-power PRA. The internal flooding portion of the PRA was not available for peer review at that time. All SRs were reviewed to the level of CC II.

A focused scope peer review of the PINGP Internal Flooding Events Model (Revision 0) against RG 1.200, Revision 2, and the ASME/ANS PRA Standard was performed in September 2012 (Reference 15). This peer review assessed all of the high level requirements and SRs in Part 3 of the ASME/ANS PRA Standard against CC II requirements.

A second focused scope peer review of the PINGP internal events PRA was conducted in April 2014 (Reference 16) to review the Flowserve N-9000 Abeyance Reactor Coolant Pump (RCP) Seal Loss of Coolant Accident modeling against RG 1.200, Revision 2, and the

ASME/ANS PRA Standard. The peer review encompassed the technical elements for accident sequence analysis, success criteria analysis, systems analysis, quantification, and LERF analysis that are impacted by the incorporation and quantification of the models for the RCP seals. This peer review was also performed against the CC II requirement of each of the applicable SRs.

The ASME/ANS PRA Standard has 326 individual SRs for the Internal Events At-Power PRA (Part 2), and Internal Flood At-Power PRA (Part 3). Collectively, the three PINGP internal events peer reviews addressed all of these SRs. Two of the SRs were judged to be not applicable. Of the remaining 324 ASME/ANS PRA Standard SRs, 97.8% were determined to be supportive of CC II or greater. A total of 31 finding-level F&Os were generated by the peer review teams, indicating areas where improvements were needed to be made to meet CC II for the remaining SRs. Subsequent to these peer reviews, NSPM implemented PRA model and documentation changes to address these F&Os.

An F&O closure review was conducted in October 2017 in accordance with the process documented in Appendix X to NEI 05-04/07-12/12-06 (Reference 17), as well as the requirements published in the ASME/ANS PRA Standard and RG 1.200, Revision 2. The findings closure review was performed by ENERCON Services, Inc. (Reference 11) and determined that 29 of the 31 findings had been closed.

A second F&O closure review was performed by ENERCON Services, Inc. in May 2019 to review further PRA model changes made in the current Revision 5.3 of the PRA (Reference 18) to address one of the two open internal events PRA F&Os, as well as open fire PRA F&Os. This second review (Reference 13) was conducted in the same manner as the first closure review, in accordance with Appendix X to NEI 05-04/07-12/12-06 requirements. Following the second closure review, one finding remains open. Table E2-1 discusses the disposition of that finding.

With the disposition of the single open peer review finding, the Revision 5.3 internal events PRA model of record meets the requirements for PRA technical adequacy for this application.

5.0 SCOPE AND TECHNICAL ADEQUACY OF PINGP FIRE PRA

The information provided in this section demonstrates that the PINGP fire PRA model meets the expectations for PRA scope and technical adequacy as presented in RG 1.200, Revision 2, to fully support the requirements of the RICT Program.

A state-of-the-art fire PRA was developed using the guidance provided by NUREG/CR-6850, “Fire PRA Methodology for Nuclear Power Facilities” (References 19 and 20), to support a LAR (Reference 21) for PINGP Fire Protection Program conversion from Appendix R of 10 CFR 50 to compliance to National Fire Protection Association (NFPA) Standard 805, “Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants”. The technical adequacy of this PRA was reviewed by NRC as part of the PINGP NFPA 805 LAR approval process (Reference 22). The fire PRA is built upon the internal events PRA which was modified to capture the effects of fire. The model is based on the

PINGP plant configuration assuming completion of NFPA 805 modifications, as detailed in 2.C.(4)(c), “Transition License Conditions”, of the Renewed Facility Operating Licenses DPR-42 and DPR-60. The current version of the fire PRA model is Revision 5.3 (Reference 23).

A full-scope fire PRA peer review was performed in June 2012 on the Revision 0 model, applying the NEI 07-12 process, the ASME/ANS PRA Standard, and RG 1.200, Revision 2. The purpose of this review was to establish the technical adequacy of the fire PRA for the spectrum of potential risk-informed plant licensing applications for which the fire PRA may be used. The 2012 PINGP fire PRA peer review was a full-scope review of all of the technical elements of the PINGP at-power fire PRA against all technical elements in Section 4 of the ASME/ANS PRA Standard, including the referenced internal events SRs in Section 2. All SRs were reviewed against the CC II requirements.

The fire PRA Section of the ASME/ANS PRA Standard has 183 individual SRs and references various SRs in the internal events PRA section; the PINGP fire PRA peer review (Reference 24) included all of the SRs and all applicable internal events SRs. For the assessment of the reviewed ASME/ANS PRA Standard SRs, 56 unique F&Os were generated by the peer review team, 40 were peer review findings, 15 were suggestions, and one was considered a best practice. There were no “unreviewed analysis methods” identified during the review.

In Revision 1.0 of the PINGP fire PRA model, the model was upgraded to a new method for determining resultant Hot Gas Layer temperature levels. To support the incorporation of this method, a focused-scope peer review was performed in March 2014 on Revision 3.0 of the fire PRA model against the CC II requirements for SRs FSS-C2, FSS-C3, FSS-C5, FSS-D1, FSS-D2, FSS-D3, FSS-D6, FSS-G1, FSS-G2, FSS-H3, FSS-H5, and FSS-H9 (Reference 25). This focused-scope peer review resulted in one additional “finding” F&O. Therefore, the Revision 3.0 model had a total of 41 open finding F&Os as a result of the two peer reviews.

A findings closure review was conducted in October 2017 in accordance with the process documented in Appendix X to NEI 07-12, as well as the requirements published in the ASME/ANS PRA Standard and RG 1.200, Revision 2. The findings closure review was performed by ENERCON Services, Inc. (Reference 10) and determined that 35 of the 41 findings were closed. A second focused scope fire PRA peer review (Reference 26) was completed in January 2018 to assess the closure of an F&O (FSS-B2-01 from the 2012 full-scope peer review) that could not be closed by the findings closure review since it was classified as a PRA upgrade. The focused scope peer review determined that the issue associated with the F&O has been addressed and the underlying SRs are met at least the CC II level.

As noted in Section 4 of this enclosure, a second F&O closure review was performed by ENERCON Services, Inc. in May 2019 (Reference 13) to review further fire PRA model changes made in the current Revision 5.3 of the fire PRA to address the open fire PRA F&Os. That closure review determined that all five F&Os were now closed in a manner that meets the CC II requirements of the SRs referenced in those F&Os.

Therefore, the Revision 5.3 fire PRA model of record meets the requirements for PRA technical adequacy for this application.

Table E2-1: PINGP Open PRA Peer Review Findings

F&O Number	SR	Peer Review Finding	Resolution	Impact on Application
Internal Events PRA Open Findings				
Finding SY-A17-01	SY-A17	From 2014 Focused Scope Peer Review for Flowserve N-9000 Reactor Coolant Pump Seal Loss of Coolant Accident (LOCA) Modeling: Subsection 1.8.1 of AC System notebook, “PRA-PI-SY-AC, Rev. 2.1a” indicates safeguards 4kV buses do not result in RCP trip. Failure in both 4kV buses (Bus 15 and 16), which is a cause of 1AC, requires RCP trip to prevent RCP seal failure, but 1N9-SBO gate does not include the operator action. Cause(s) of loss of 1AC which do not result in RCP trip requires RCP trip within 2 hours to prevent an RCP LOCA.	A sensitivity analysis was performed that demonstrated failure to trip the RCP with a loss of all 4kV safety buses would result in a negligible (less than 1E-8/year) increase in CDF and represents a negligible source of uncertainty for the base PRA model. The F&O finding closure review team concurred with this assessment. However, since the N-9000 RCP seal model must obtain NRC review and approval, the closure review team determined that the F&O should remain open until the underlying RCP seal model is approved.	It is expected that this F&O finding can be considered closed once the underlying RCP seal model has been approved. An additional sensitivity calculation was performed for this application. This sensitivity re-calculated all RICTs from Enclosure 1 of this LAR with the abeyance credit removed. The results of this study showed that all but one of the cases evaluated resulted in RICTs with the same duration in days, except for one RICT that decreased by 1 day. This sensitivity shows that the impact of the Abeyance Seal on this application is minimal.

6.0 REFERENCES

1. Letter from the Technical Specification Task Force (TSTF) to the NRC, “TSTF Comments on Draft Safety Evaluation for Traveler TSTF-505, ‘Provide Risk-Informed

Extended Completion Times' and Submittal of TSTF-505, Revision 2", Revision 2, dated July 2, 2018 (ADAMS Accession No. ML18183A493)

2. Letter (L-PI-18-012) from NSPM to the NRC, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors'", dated July 20, 2018 (ADAMS Accession No. ML18204A393)

3. Letter (L-PI-18-005) from NSPM to the NRC, "License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805", dated May 18, 2018 (ADAMS Accession No. ML18138A402)

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

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

6. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 2, dated May 2011 (ADAMS Accession No. ML100910006)

7. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 2, dated March 2009 (ADAMS Accession No. ML090410014)

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

9. NEI Topical Report NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard", Revision 2, dated November 2008 (ADAMS Accession No. ML083430462)

10. NEI Topical Report NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines", Revision 1, dated June 2010 (ADAMS Accession No. ML102230070)

11. NSPM PRA Document V.SPA.17.010, "Prairie Island Internal Events Probabilistic Risk Assessment Peer Review Findings Closure", Revision 0, dated November 2017

12. NSPM PRA Document V.SPA.17.009, "PINGP Fire PRA Closure Report", Revision 0, dated November 2017

13. NSPM PRA Document V.SPA.19.007, “Prairie Island Internal Events and Fire Probabilistic Risk Assessment Peer Review Findings Closure”, Revision 0, dated May 2019

14. NSPM PRA Document V.SPA.10.020, “RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirements For the Prairie Island Nuclear Generating Plant Probabilistic Risk Assessment (Internal Events, No Internal Flooding)”, Revision 0, dated September 2017

15. NSPM PRA Document V.SPA.12.036, “Focused Scope RG 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements For the Prairie Island Nuclear Generating Plant Internal Flood Probabilistic Risk Assessment”, Revision 0, dated September 2017

16. NSPM PRA Document V.SPA.14.018, “Focused Scope PRA Peer Review of the Prairie Island Nuclear Generating Plant Probabilistic Risk Assessment against the PRA Standard Supporting Requirements From Section 2 of the ASME/ANS Standard”, Revision 0, dated September 2017

17. NEI TR 05-04/07-12/12-06 Appendix X, “NEI 05-04/07-12/12-06 Appendix X: Close Out of F&Os”, dated June 2016 (ADAMS Accession No. ML16158A035)

18. NSPM PRA Document PRA-PI-QU, “PRA Level 1 Quantification”, Revision 5.3, dated November 2017

19. NRC NUREG/CR-6850 Volume 2, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology”, dated September 2005 (ADAMS Accession No. ML15167A411)

20. NRC NUREG/CR-6850, Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancements”, dated September 2010 (ADAMS Accession No. ML103090242)

21. Letter (L-PI-12-089) from NSPM to NRC, “License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors”, dated September 28, 2012 (ADAMS Accession No. ML12278A405)

22. Letter from the NRC to NSPM, “Prairie Island Nuclear Generating Plant, Units 1 and 2 - Issuance of Amendments Re: Transition to NFPA-805 ‘Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants’ (CAC Nos. ME9734 and ME9735)”, dated August 8, 2017 (ADAMS Accession No. ML17163A027)

23. NSPM PRA Document FPRA-PI-FQ, “Fire PRA Quantification Notebook”, Revision 5.3, dated April 2018

24. NSPM PRA Document V.SPA.12.037, “Fire PRA Peer Review", Revision 0, dated September 2017

25. NSPM PRA Document V.SPA.13.014, “Fire PRA Focused Scope Peer Review on Hot Gas Layer”, Revision 0, dated September 2017

26. NSPM PRA Document V.SPA.18.002, “PINGP FPRA Focused-Scope Peer Review” Revision 0, dated January 2018

ENCLOSURE 3

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

INFORMATION SUPPORTING TECHNICAL ADEQUACY OF PRA MODELS WITHOUT PRA STANDARDS ENDORSED BY REGULATORY GUIDE 1.200, REVISION 2

(1 Page Follows)

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

This enclosure is not applicable to the Prairie Island Nuclear Generating Plant submittal. Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy, is not proposing to use any PRA models in the PINGP Risk-Informed Completion Time Program for which a PRA standard endorsed by the NRC in Regulatory Guide 1.200, Revision 2, does not exist.

ENCLOSURE 4

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

INFORMATION SUPPORTING JUSTIFICATION OF EXCLUDING SOURCES OF RISK NOT ADDRESSED BY THE PRA MODELS

(19 Pages Follow)

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

1.0 INTRODUCTION AND SCOPE

Nuclear Energy Institute (NEI) Topical Report NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 1), as clarified by the NRC final safety evaluation (Reference 2), requires that the license amendment request (LAR) provide a justification for exclusion of risk sources from the Probabilistic Risk Assessment (PRA) model based on their insignificance to the calculation of configuration risk, and to discuss conservative or bounding analyses applied to the configuration risk calculation. This enclosure addresses this requirement by discussing the overall generic methodology to identify and disposition such risk sources, and providing the Prairie Island Nuclear Generating Plant (PINGP)-specific results of the application of the generic methodology and the disposition of impacts on the PINGP Risk-Informed Completion Time (RICT) Program. Section 3.0 of this enclosure presents the plant-specific bounding analysis of seismic risk to PINGP. Section 4.0 presents the justification for excluding analysis of other external hazards from the PINGP PRA. The PINGP internal events (including internal flooding) and fire PRA models described within this LAR are the same as those described within Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), submittals regarding adoption of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (Reference 3), and modification of the list of required NFPA 805 modifications (Reference 4), respectively.

NEI 06-09-A does not provide a specific list of hazards to be considered in a RICT Program. However, non-mandatory Appendix 6-A of the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) RA-Sa-2009 PRA Standard (hereafter “ASME/ANS PRA Standard”), “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Reference 5) provides a guide for identification of most of the possible external events for a plant site. Additionally, NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, Revision 1 (Reference 6), provides a discussion of hazards that should be evaluated to assess uncertainties in plant PRAs and support the risk-informed decision-making process. These hazards were reviewed for PINGP, along with a review of information pertaining to the site region and plant design to identify the set of external events to be considered. Information from the PINGP Updated Safety Analysis Report (USAR) pertaining to the geologic, seismologic, hydrologic, and meteorological characteristics of the site region, and the current and projected industrial activities in the plant vicinity was reviewed. No new site-specific or plant-unique external hazards were identified through this review. The list of hazards from Appendix 6-A of the ASME/ANS PRA Standard that were considered for PINGP is summarized in Table E4-2.

The scope of this enclosure is consideration of the hazards listed in Table E4-2 for applicability to PINGP. Seismic events in particular are evaluated quantitatively in Section 3.0, and the other listed external hazards are evaluated and screened as low risk in Section 4.0.

2.0 TECHNICAL APPROACH

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

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

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

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

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

2.1 Hazard Screening

The first step in the evaluation of the external hazard is screening based on an estimation of a bounding core damage frequency (CDF) for beyond design basis hazard conditions. An example of this type of screening is reliance on the NRC's 1975 Standard Review Plan (SRP) (Reference 7) which is acknowledged in the NRC's Individual Plant Examination of External Events (IPEEE) procedural guidance (Reference 8) as assuring a bounding CDF of less than 1E-06 per year for each hazard. The bounding CDF estimate is often characterized by the likelihood of the site being exposed to conditions that are beyond the design basis limits and an estimate of the bounding conditional core damage probability for those conditions. If the bounding CDF for the hazard can be shown to be less than 1E-06 per year, then beyond design basis challenges from the hazard can be screened and do not need to be addressed quantitatively in the RICT Program. The basis for this is as follows:

- The overall calculation of the RICT is limited to an incremental core damage probability (ICDP) of 1E-05.
- The maximum time interval allowed for the RICT is 30 days.
- If the maximum CDF contribution from a hazard is <1E-06 per year, then the maximum ICDP from the hazard is <1E-07 (1E-06/year * 30 days/365 days/year).
- Thus, the bounding ICDP contribution from the hazard is shown to be less than 1% of the permissible ICDP in the bounding time for the condition. Such a minimal contribution is not significant to the decision in computing a RICT.

The PINGP hazard screening analysis from the IPEEE has been updated to reflect current site conditions. The results are discussed in Section 4.0, and show that all events listed in Table E4-2 can be screened for PINGP, except for seismic events.

While the direct CDF contribution from beyond design basis hazard conditions can be shown to be non-significant using this approach, some external hazards can cause a plant challenge even for hazard severities that are less than the design basis limit. These considerations are addressed in Section 4.0.

2.2 Hazard Analysis for CDF Contribution

There are two options in cases where the bounding CDF for the external hazard cannot be shown to be less than 1E-06 per year. The first option is to develop a PRA model that explicitly models the challenges created by the hazard and the role of the SSCs included in the RICT Program in mitigating those challenges. The second option for addressing an external hazard is to compute a bounding CDF contribution from the hazard. The bounding approach used for seismic risk is described in Section 3.0.

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

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

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

The bounding approach used for seismic LERF is described in Section 3.0.

2.4 Risks from Hazard Challenges

Upon estimation of a bounding CDF and LERF, the analysis approach must assure that the RICT Program calculations reflect the change in CDF and LERF caused by out-of-service equipment. As discussed in Section 3.0, seismic risk is the only beyond design basis hazard that could not be screened out for PINGP. The approach used considers that the change in risk with equipment out of service will not be higher than the bounding seismic CDF.

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

The approach to be taken in this step is to identify the plant challenges caused by the occurrence of the hazard within the design basis and evaluate whether the risks associated with these events are either already considered in the existing PRA model or they not significant to the risk. Section 3.0 provides the analysis of the beyond design basis seismic hazards for the PINGP site, and Section 4.0 provides an analysis of the representative external hazards for PINGP.

3.0 SEISMIC BOUNDING ANALYSIS

This section presents the analysis that bounds the potential seismic impact for inclusion in the decision-making process, as a seismic PRA (SPRA) is not available for PINGP. The process for analyzing an unscreened external hazard without the use of a full PRA involves the following three steps:

1. Estimate Bounding CDF
2. Evaluate Potential Risk Increases Due to Out of Service Equipment
3. Qualitatively Evaluate Bounding LERF Contribution

3.1 Estimate Bounding Seismic CDF

A seismic PRA is not developed for PINGP. NSPM performed the equivalent of a reduced-scope seismic margins assessment (SMA) for the PINGP IPEEE (Reference 9), with an additional focus on a few components, in accordance with Supplement 5 of Generic Letter 88-20 (Reference 10). The seismic hazard for the PINGP site was re-evaluated in 2014 and provided to the NRC (Reference 11). The site safe shutdown earthquake (SSE) is documented in this report as 0.12 g. For screening purposes, a Ground Motion Response Spectrum (GMRS) was developed and a probabilistic seismic hazard analysis was completed using the

Central and Eastern United States (CEUS) Seismic Source Characterization for nuclear facilities and the updated Electric Power Research Institute (EPRI) Ground-Motion Model. For both the 1 to 10 Hz response spectrum and higher frequency (>10 Hz), the SSE bounds the GMRS, therefore no further evaluation was performed. The NRC concurred that the reevaluated seismic hazard is bounded by the plants existing design-basis SSE and that no further responses or regulatory actions associated with Phase 2 of Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic" were required for PINGP (Reference 12).

Therefore, an alternative approach is taken to provide an estimate of seismic core damage frequency (SCDF) based on the current PINGP seismic hazard curve and assuming the seismic capacity of a component whose seismic failure would lead directly to core damage. This approach to estimation of the SCDF uses the plant level high confidence of low probability of failure (HCLPF) seismic capacity obtained from Table C-2 of Reference 13 and convolves the corresponding failure probabilities as a function of seismic hazard level with the seismic hazard curve from Reference 11. This is a commonly used approach to estimate SCDF when a seismic PRA is not available; see Section 10-B.9 of the ASME/ANS PRA Standard. This approach is consistent with approaches that have been used in other regulatory applications.

The EPRI completed site-specific evaluations using new site-specific hazard estimates for plants in the CEUS (Reference 14). The Expedited Seismic Evaluation Process (ESEP) for sites with a GMRS that exceeds the SSE in the spectral range from 1 to 10 Hz developed SCDF estimates and compared them to the SCDF estimates previously developed by the NRC (Reference 13). The approach in the 2014 EPRI evaluation estimated SCDF using the plant-level HCLPF seismic capacity (0.28g), composite variability (βc of 0.4), and spectral ratios for PINGP from Table C-2 of Reference 13, convolved with the new site-specific seismic hazard. This approximation is consistent with the approach and calculated SCDF values from Appendix D of Reference 13, which ranged from 1.4E-06 to 6.0E-06 for PINGP. Using the same PINGP HCLPF and spectral ratios, and the hazard curves from Reference 11, the total PINGP SCDF is estimated to be 3.0E-06. This SCDF value will be used as the bounding estimate of instantaneous SCDF ($ICDF_{seismic}$) for the PINGP TSTF-505 LAR RICT calculations.

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

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

If the industry accepted approach (Reference 13) of correlation is assumed, the conditional core damage frequency given a seismic event will remain unaltered whether equipment is out of service or not. Thus, the risk increase due to out of service equipment cannot be greater than the total SCDF estimated by the bounding method used in Reference 14. That is, for the PINGP site, the delta SCDF from equipment out of service cannot be greater than 3.0E-06 per year.

To summarize the above considerations:

- The baseline seismic risk in this approach is assumed to be zero, whereas there will always be some level of baseline seismic risk for a zero-maintenance plant configuration. Therefore, the incremental seismic risk (configuration seismic risk – baseline seismic risk) will always be overstated using a seismic penalty based on the total estimated seismic risk.

- The limiting HCLPF approach assumes that a failure of a component with seismic capacity at that HCLPF leads directly to core damage (CD). However, even common failure of a given set of components (e.g., all emergency diesel generators (DGs)) would not lead directly to CD, especially in light of the post-Fukushima FLEX mitigating strategies now in place. In reality, there are few SSCs whose failure would lead to seismic CD with any significant frequency. Examples could be important structures, or the reactor pressure vessel, or “distributed systems”, such as all cable trays or all piping systems.

- In a seismic PRA, seismic impacts to similar components (e.g., all the DGs for a given unit) are typically assumed to be correlated unless there are reasons to justify not correlating. Correlation has the effect of introducing common cause impacts. So, if one train of emergency AC power fails seismically, both trains are modeled as likely to fail given the same seismic event. So, in general, most seismic impacts would effectively be equivalent to TS loss of function.

- Given the above, the use of a seismic penalty based on assuming seismic core damage given the plant level HCLPF is appropriate.

Note that there is another significant conservatism inherent in this approach in addition to the above considerations. In determining the SCDF to be used in the RICT calculations, the full annual seismic hazard has been used. Since the maximum RICT backstop is 30 days, accounting for the full hazard introduces more than a factor of 10 increase in the calculated SCDF.

3.3 Evaluate Seismic LERF Contribution

The current PINGP full-power internal events (FPIE) PRA (Reference 15) includes a comprehensive treatment of LERF due to internally-initiated events. The internal events PRA provides an estimate of the conditional probability of LERF for each modeled initiating event. Seismic events would not be expected to induce containment bypass scenarios (e.g., Interfacing Systems Loss of Coolant Accident (ISLOCA) or Steam Generator Tube Rupture (SGTR), and the bypass resulting from ISLOCA or SGTR is not a function of containment seismic capability. Therefore, a bounding conditional large early release probability for seismic events ($CLERP_{seismic}$) can be obtained by examining the event-specific CDF and event-specific LERF, for the non-direct bypass events:

$$CLERP_{IE} = LERF_{IE} / CDF_{IE}$$

Using the current PINGP FPIE PRA, the average CLERP over all initiating events other than direct containment bypass events is approximately 1.0% for both units as shown in Table E4-1 below:

Table E4-1: PINGP Non-Bypass CLERF Summary

Unit	LERF (per reactor critical years (/ RCY))	Non-Bypass LERF (/ RCY)	CDF (/ RCY)	Non-Bypass CDF (/ RCY)	Non-Bypass CLERP
1	2.15E-07	1.16E-07	1.28E-05	1.12E-05	1.0%
2	1.86E-07	1.10E-07	1.25E-05	1.08E-05	1.0%

The CLERP for the PINGP initiating events ranges from 0% to 4.1% for both units. A LERF-weighted average CLERP can be computed for each initiating event as follows:

$$CLERP_{weighted\ event\ i} = CLERP_{event\ i} * (LERF_{event\ i} / \text{Total LERF})$$

The overall weighted CLERP is the sum of the event CLERP values. The weighted CLERP calculated for the PINGP FPIE model results other than direct containment bypass events, including the events with CLERP values above the average CLERP, is 0.55% for Unit 1 and 0.61% for Unit 2.

Only one, non-bypass initiating event results in CLERP values greater than 1.2% (small loss of coolant accident has a CLERP value of 4.1% for Unit 1 and 3.8% for Unit 2). The event-specific CLERP for all other initiating events is less than 1.2%. Based on the above discussion, a 5% value of CLERP is chosen as an adequately conservative, but not overly pessimistic, estimate for use in the seismic induced LERF calculation. This encompasses all internal events initiators contributing to total LERF and total CDF for those events that do not result in direct containment bypass.

The incremental bounding large early release frequency from seismic events (i.e., the SLERF) for use in RICT calculations is then computed as:

$$ILERF_{Seismic} = ICDF_{Seismic} * CLERP_{Seismic} = 3.0E\text{-}06 * 0.05 = 1.5E\text{-}07$$

Since this estimation of CLERP may change as the internal events PRA model is updated, the estimate will be updated for the RICT Program with each internal events model update.

3.4 Conclusion

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

The RICT and RMAT calculations are based on the discussion provided above. The actual RICT and RMAT calculations performed by the PINGP Configuration Risk Management Tool are based on adding an incremental 3.0E-06 per year seismic CDF contribution and a corresponding 1.5E-07 per year seismic LERF contribution to the configuration-specific delta CDF and delta LERF attributed to internal and fire events contributions. This is accomplished by adding these seismic contributions to the instantaneous CDF/LERF whenever a RICT is in effect. This method ensures that an incremental seismic CDF/LERF equal to the bounding SCDF/SLERF is added to internal and fire events incremental CDF/LERF contribution for every RICT occurrence.

4.0 EVALUATION OF EXTERNAL EVENT CHALLENGES AND IPEEE UPDATE RESULTS

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

4.1 Hazard Screening Except Seismic Events

The PINGP IPEEE for Units 1 and 2 (Reference 9) provides an assessment of the risk to the PINGP associated with external hazards. Additional analyses have been done since the IPEEE to provide updated risk assessments of various hazards, such as aircraft impacts, industrial facilities and pipelines, and external flooding (Reference 16).

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

External hazards other than seismic can be screened for the PINGP site.

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

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

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

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

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

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[a]	Disposition for RICT
Aircraft Impact	Y	PS4	The nearest major airport is Minneapolis-St. Paul International (MSP) which is located approximately 30 miles from the site. There are only three airports within 10 miles of the plant and they have been screened out from further consideration or analyzed to be so small as to not pose a hazard for PINGP. Of the airports greater than 10 statute miles from the PINGP site, they have either been screened out or demonstrated to not pose a hazard for PINGP. A reevaluation of external events demonstrated that the risk due to this hazard of aircraft-induced radiological consequences is less than 1.0E-07 per year. If it is conservatively assumed that LERF is a surrogate for the radiological consequence and CDF is typically an order of magnitude greater for PWRs, this would imply that CDF is less than 1.0E-06 per year, which satisfies Criterion PS4. Therefore, this

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[(a)]	Disposition for RICT
			hazard can be excluded from the RICT Program evaluation.
Avalanche	Y	C3	The topography surrounding PINGP precludes the possibility of a snow avalanche.
Biological Event	Y	C5	Actions committed to and completed by PINGP in response to GL 89-13 (Service Water System Problems Affecting Safety-Related Equipment) provide on-going control of biological hazards. These controls are described in PINGP procedure H21, “Generic Letter 89-13 Implementing Program”. Additionally, actions taken in response to INPO SOER 07-2 (Intake Structure Blockage) provide an additional layer of biological hazard management. The hazard is slow to develop and can be identified by monitoring and managed through standard maintenance processes.
Coastal Erosion	Y	C3	The mid-western inland location of PINGP precludes the possibility of coastal erosion.
Drought	Y	C1 C5	These effects would take place slowly allowing time for orderly plant reductions including shutdowns. Also, the design of the cooling water supply is such that adequate water will be delivered into the plant under any condition.
External Flooding and Intense Precipitation	Y	PS1	The external flooding hazard at PINGP was recently updated as a result of the post-Fukushima 50.54(f) Request for Information and the flood hazard reevaluation report (FHRR) was submitted to the NRC for review on May 9, 2016 (Reference 17). The results indicate that flooding from rivers and streams are bounded by the current licensing basis and do not pose a challenge to the plant. Flooding from local intense precipitation (LIP) was also evaluated and the focused evaluation (Reference 18) affirms that during LIP events the site has effective flood protection through the determination of Available Physical Margin and the reliability of protection features and will not challenge any safety functions at PINGP. Therefore, this hazard can be excluded from the RICT Program evaluation.
Extreme Wind or	Y	PS2	Wind damage is bounded by tornadoes, and the

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[(a)]	Disposition for RICT
Tornadoes		PS4	tornado wind speed corresponding to the 1.0E-07 per year exceedance frequency is less than the PINGP design value. Therefore, damage due to the forces associated with extreme winds or tornadoes, including missiles, can be excluded from the RICT Program evaluation.
Fog	Y	C4	The principal effects of such events would be to indirectly cause a loss of offsite power due to the occurrence of other hazards, such as highway accidents, aircraft landing and take-off accidents, and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
Forest or Range Fire	Y	C1 C3 C4	The site landscaping and lack of forestation nearby prevent such fires from posing a threat to PINGP. Furthermore, the principal effects of such events would be to cause a loss of offsite power, which is addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP, and smoke and gases entering the control room. If the latter were to occur, operators would have sufficient time to take action, such as donning protective air masks within the control room if the concentration of smoke begins to increase.
Frost	Y	C4	The effects of frost are bounded by snow and ice. The principal effect of such events would be to cause a loss of offsite power and is addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
Hail	Y	C1 C4	Hail is bounded by other events, such as tornado missiles, for which the plant is designed. The principal effects of such events would be to cause a loss of offsite power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
High Summer Temperature	Y	C1 C5	The principal effects of such events would result in elevated river temperatures which are monitored by station personnel. The design maximum temperature for the Cooling Water System is 85°F and the average monthly temperature at St. Paul, which is typically 2 to 3 degrees higher than at the site, typically does not approach that value. Safeguards components are operable with Cooling Water inlet

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[(a)]	Disposition for RICT
			temperature up to 95°F. The climatology at PINGP is such that extreme heat would have an insignificant effect on plant operations.
High Tide, Lake Level, or River Stage	Y	C3 C4	High tide or lake level are not applicable to the site because of location. Impact of High River Stage is included as an impact in the external flooding analysis.
Hurricane	Y	C3	The mid-western location of PINGP precludes the possibility of a hurricane. Additionally, hurricanes would be covered under Extreme Winds and Tornados and Local Intense Precipitation.
Ice Cover	Y	C1 C4	Plant piping and equipment located outside of plant buildings are protected by heat tracing to prevent adverse effects from severe cold. Furthermore, the capacity reduction of the ultimate heat sink (UHS) due to extreme cold would be a slow process that would allow plant operators sufficient time to take proper actions, such as reducing plant power output level or achieving safe shutdown. The principal effects of such events would be to cause a loss of offsite power and are addressed in the weather-related Loss of Offsite Power initiating event in the internal events PRA model for PINGP.
Industrial or Military Facility Accident	Y	C3 C4	There are no military facilities within five miles of the plant (the closest is the Red Wing National Guard Armory, ~7.5 miles away). The hazards associated with an industrial facility accident are screened elsewhere in this table (e.g., transportation accident, pipeline accident).
Internal Fire	N	None	The PINGP NFPA 805 fire PRA addresses risk from internal fire events.
Internal Flooding	N	None	The PINGP internal events PRA addresses risk from internal flooding events.
Landslide	Y	C3	In the immediate vicinity of the PINGP, there are no steep hills. Therefore, it is not applicable to the site because of topography.
Lightning	Y	C1 C4	Lightning strikes can result in loss of offsite power. This is incorporated into the PINGP internal events PRA model through the incorporation of generic and plant-specific data.

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[a]	Disposition for RICT
Low Lake Level or River Stage	Y	C1 C5	PINGP uses water from the reservoir upstream of Lock and Dam Number 3 on the Mississippi River for UHS. An accident at the dam concurrent with normal river flow would provide a level of water 10.9 feet deep at the circulating water intake. The design of the cooling water system is such that it will deliver adequate water to the plant under any condition. Other reductions in river level would take place slowly over time allowing for orderly plant reductions, including shutdowns.
Low Winter Temperature	Y	C1 C4 C5	Plant piping and equipment located outside of plant buildings are protected by heat tracing to prevent adverse effects from severe cold. The principal effects of such events would be to cause a loss of offsite power. The effects of weather-related losses of offsite power are included in the PINGP PRA models. These effects would take place slowly allowing time for orderly plant power reductions, including shutdowns.
Meteorite/Satellite Strike	Y	C2	The frequency of a meteorite or satellite strike is judged to be very low such that the risk impact from such events is insignificant.
Pipeline Accident	Y	C1	A 4-inch natural gas supply line terminates outside the northwest corner of the Owner Controlled Area for PINGP. The effects on plant structures due to a blast release due to a Vapor Cloud Explosion are bounded by tornado loadings.
Release of Chemicals from On-site storage	Y	C3 C4 PS1	No chlorine gas is stored on-site. The newly installed natural gas supply line was also evaluated for its effect on control room habitability and diesel generator operation where it was determined that natural gas concentrations resulting from a leak would neither challenge the environment of the control room nor would it challenge the operability of the safety-related diesel generators. In addition, it was determined there was no asphyxiation hazard posed by the rupture of the largest nitrogen tank onsite (3000 gallons). Chemical hazards stored and transported in the vicinity of the plant are analyzed in conformance with the guidance set forth by RG 1.78 and NUREG-0570. Therefore, this hazard can be excluded from the RICT Program evaluation.
River Diversion	Y	C1	In the event of Mississippi River diversion, the water

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[(a)]	Disposition for RICT
			in the intake canal and the emergency intake line provide enough cooling water to enable safe shutdown of both units.
Sand or Dust Storm	Y	C1 C3 C4	The frequency of a loss of offsite power accounts for severe weather, including sand or dust storms.
Seiche	Y	C3	The PINGP site is located on the Mississippi River. Gantenbein Lake and Larson Lake are both more than 1/2 of a mile from the site and Sturgeon Lake is approximately 1/3 of a mile from the site. Therefore, no large body of water is close enough for the site to be susceptible to a seiche.
Seismic Activity	N	None	Seismic impacts are evaluated in terms of a bounding SCDF applied to the calculation of RICT values. See section 3 of this enclosure.
Snow	Y	C1 C4	The average snowfall per year in Red Wing, Minnesota is 32 inches. The maximum recorded snowfall from a single storm in Minnesota occurred near Finland and measured 46.5 inches. One inch of snowfall weighs approximately 1 psf, which means the estimated weight from a postulated maximum snowfall would be 46.5 psf. The design basis roof live load is 50 psf, which is within the design basis.
Soil Shrink-Swell	Y	C3	The soil at the site is sandy alluvium. Due to the very permeable nature of the granular soils at the site, the soil is resistant to shrink-swell.
Storm Surge	Y	C4	The potential storm surge from Sturgeon Lake was evaluated in the FHRR and determined to be bounded by External Flooding.
Toxic Gas	Y	C4	The hazards associated with toxic gas are screened elsewhere in this table (e.g., Industrial and Military Facility Accidents, Release of Chemicals in Onsite Storage).
Transportation Accidents	Y	C1 C3 C4	Land Transportation – Based on the proximity of the nearest major roadways, truck explosions pose no danger to PINGP and the impact of toxic gas release has been evaluated and shown to be negligible. Rail Transportation – Based on the proximity of the nearest commercial railroad line, potential impacts of explosions are covered by Extreme Wind or Tornado and the impact of toxic gas release has been

Table E4-2: Evaluation of Risks from External Hazards (Reference 16)

External Hazard	Screened Out? (Y/N)	Screening Criterion[a]	Disposition for RICT
			evaluated and shown to be negligible. Water Transportation – PINGP is located along the Mississippi River, the main channel of which is ~0.5 miles from the site. Based on that proximity, potential impacts of explosions are covered by Extreme Wind or Tornado and the impact of toxic gas release has been evaluated and shown to be negligible.
Tsunami	Y	C3	The mid-western location of PINGP precludes the possibility of a tsunami.
Turbine-Generated Missiles	Y	PS4	The probabilistic analysis performed for postulated failures of turbines in PINGP has shown that the overall probability of turbine missile damage is less than the NRC-accepted value of 1.0E-07 per year. Therefore, this hazard can be excluded from the RICT Program evaluation.
Volcanic Activity	Y	C3	Not applicable to PINGP as the site is not close to any active volcanoes.
Waves	Y	C4	The potential impacts of waves were evaluated in the FHRR and determined to be bounded by External Flooding.

Note (a): See Table E4-3 for descriptions of screening criteria.

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

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is less than events for which plant is designed.	NUREG/CR-2300 (Reference 19) ASME/ANS Standard RA-Sa-2009 (Reference 5)	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 ASME/ANS Standard RA-Sa-2009	

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

Event Analysis	Criterion	Source	Comments
	C4. Event is included in the definition of another event.	NUREG/CR-2300 ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP) (Reference 8).	NUREG-1407 (Reference 8) ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-05 per year and the mean conditional core damage probability is < 0.1.	NUREG-1407 ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-06 per year.	NUREG-1407 ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ ANS PRA Standard.	NUREG-1407 ASME/ANS Standard RA-Sa-2009	

4.2 Seismically-Induced Loss of Offsite Power Challenges

For the PINGP site, the only incremental risk associated with challenges to the facility that do not exceed the design capacity that is not already addressed is seismically-induced loss of offsite power (LOOP). The Risk Assessment of Operational Events Handbook (Reference 20) presents a calculation of the frequency for seismically-induced LOOP events for all U.S. nuclear power plants, based on the lowest fragility SSCs (e.g., ceramic insulators). The seismic initiating event frequency used in Reference 20 was obtained from the PINGP seismic hazard distribution developed in response to NTTF Recommendation 2.1 (References 11 and 12).

As obtained from Table A-0-1 of Reference 20, the seismic-induced LOOP frequency for PINGP is 3.25E-06 per year. The internal events PRA models LOOP from plant-centered, switchyard-centered, grid-related, and weather-related events. Based on the PINGP internal events PRA, total frequency of unrecovered loss of offsite power (i.e., the sum of the frequency times the non-recovery probability at 24 hours over these LOOP events) is 5.48E-05 per year (Reference 21).

The seismically-induced (unrecoverable) LOOP frequency is therefore less than 6% of the total unrecovered LOOP frequency that is already accounted for in the internal events PRA. This frequency is judged to be a sufficiently small fraction that it will not significantly impact the RICT Program calculations and it can be omitted.

5.0 CONCLUSIONS

Based on this analysis of external hazards for PINGP Units 1 and 2, no additional external hazards other than seismic events need to be added to the existing PRA model. The evaluation concluded that the hazards either do not present a design-basis challenge to PINGP, the challenge is adequately addressed in the PRA, or the hazard has a negligible impact on the calculated RICT and can be excluded.

The ICDP/ILERP acceptance criteria of 1E-05/1E-06 will be used within the RICT Program framework to calculate the resulting RICT and RMAT based on the total configuration-specific delta CDF/LERF attributed to internal events and internal fire, plus the seismic bounding delta CDF/LERF values.

6.0 REFERENCES

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

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

3. Letter (L-PI-18-012) from NSPM to the NRC, “Application to Adopt 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors’”, dated July 20, 2018 (ADAMS Accession No. ML18204A393)

4. Letter (L-PI-18-005) from NSPM to the NRC, “License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805”, dated May 18, 2018 (ADAMS Accession No. ML18138A402)

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

6. NRC NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466)

7. NRC NUREG-75/087, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition”, September 1975 (ADAMS Accession No. ML081510817)

8. NRC NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities”, dated June 1991 (ADAMS Accession No. ML063550238)

9. NSPM Report NSPLMI-96001, “Prairie Island Nuclear Generating Plant Individual Plant Examination of External Events (IPEEE), NSPLMI-96001”, Revision 1, dated September 1998

10. NRC Generic Letter GL 88-20, “Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f) (Generic Letter No. 88-20)”, dated November 23, 1988 (ADAMS Accession No. ML031150465)

11. Letter (L-PI-14-028) from NSPM to the NRC, “PINGP Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident”, dated March 27, 2014 (ADAMS Accession No. ML14086A628)

12. Letter from the NRC to NSPM, “Prairie Island Nuclear Generating Plant, Units 1 and 2 – Staff Assessment of Information Provided Pursuant to Title 10 of the Code of Federal Regulations Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force (NTTF) Review of Insights from the Fukushima Dai-ichi Accident and Staff Closure of Activities Associated with NTTF Recommendation 2.1, ‘Seismic’ (TAC Nos. MF3784 and MF3785”, dated December 15, 2015 (ADAMS Accession No. ML15341A162)

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

14. Letter from EPRI to NEI, “Fleet Seismic Core Damage Frequency Estimates for Central and Eastern U.S. Nuclear Power Plants Using New Site-Specific Seismic Hazard Estimates”, dated March 11, 2014 (ADAMS Accession No. ML14080A589)

15. NSPM PRA Document PRA-PI-LE, “LERF Notebook Limited Level 2 (LERF) PRA”, Revision 5.3, dated November 2017

16. ENERCON Calculation, XCELCORP~00007-REPT-01, “Prairie Island Re-examination of External Events Evaluation in the IPEEE”, Revision 0, dated June 4, 2018

17. NSPM letter (L-PI-16-039) to the NRC, “Prairie Island Nuclear Generating Plant, Units 1 and 2, Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 2, Flood Hazard Reevaluation Report”, dated May 9, 2016 (ADAMS Accession No. ML16133A041)

18. NSPM letter (L-PI-16-095) to the NRC, “Prairie Island Nuclear Generating Plant, Units 1 and 2, Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 3, Focused Evaluation”, dated December 13, 2016 (ADAMS Accession No. ML16351A209)

19. NRC NUREG/CR-2300, Volume 2, “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants”, January 1983 (ADAMS Accession No. ML063560440)

20. NRC Handbook, “Risk Assessment of Operational Events Handbook, Volume 2 – External Events, Internal Fires – Internal Flooding – Seismic – Other External Events – Frequencies of Seismically-Induced LOOP Events”, Revision 1.02, dated November 2017 (ADAMS Accession No. ML17349A301)

21. NSPM PRA Document PRA-PI-IE, “Initiating Events Notebook”, Revision 5.3, dated November 2017

ENCLOSURE 5

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

BASELINE CDF AND LERF

(4 Pages Follow)

Baseline CDF and LERF

1.0 INTRODUCTION

Section 4.0, Item 6 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A, Revision 0, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines” (Reference 2), requires that the license amendment request (LAR) provide the plant-specific total core damage frequency (CDF) and large early release frequency (LERF) to confirm applicability of the limits of Regulatory Guide (RG) 1.174, Revision 1 (Reference 3). (Note that RG 1.174, Revision 2 (Reference 4), issued by the NRC in May 2011, did not revise these limits.) The PINGP internal events (including internal flooding) and fire PRA models described within this LAR are the same as those described within Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), submittals regarding adoption of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (Reference 5), and modification of the list of required NFPA 805 modifications (Reference 6), respectively.

The purpose of this enclosure is to demonstrate that the Prairie Island Nuclear Generating Plant (PINGP) total CDF and total LERF are below the guidelines established in RG 1.174. RG 1.174 does not establish firm limits for total CDF and LERF, but recommends that risk-informed applications be implemented only when the total plant risk is no more than about 1E-4/year for CDF and 1E-5/year for LERF. Demonstrating that these limits are met confirms that the risk metrics of NEI-06-09-A can be applied to the PINGP Risk Informed Completion Time (RICT) Program.

2.0 TECHNICAL APPROACH

The PINGP PRA model maintenance and update process includes “model of record” updates which are full scope model updates that include all documentation required by the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009 PRA Standard (hereafter “ASME/ANS PRA Standard”), “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Reference 7) and “application specific models” which are created using the model of record as a starting point and modified to update the PRA, implement plant changes, or correct errors to support one or more risk-informed applications. The application specific models contain one or more updates to the model of record and are documented as a standalone model. As documented in Enclosure 2, the current model of record for the Full Power Internal Events (FPIE) PRA is Revision 5.3 and the model of record for the fire PRA is also Revision 5.3. In addition to the model of record, both PRA models have been revised to create an application specific model. The FPIE PRA Revision 5.3-APP1 model was created to support the Maintenance Rule (a)(4) Program. The fire PRA Revision 5.3-APP1 model was created in support of the PINGP LAR to revise the license condition associated with implementation of the transition to National Fire Protection Association (NFPA) Standard 805 (Reference 8). Both of these application-specific models were used as a starting point to support creation of sample RICT timeframes in Enclosure 1. For completeness, the baseline CDF/LERF values from both the models of record and the application-specific models are included in this enclosure.

The following tables include the PINGP Unit 1 and Unit 2 CDF and LERF values from a quantification of the applicable model revision for both FPIE (including internal flooding) and fire PRA. The tables also include an estimate of the seismic contribution to CDF and LERF, as described in Enclosure 4. Other external hazards are below accepted screening criteria and therefore do not contribute significantly to the totals.

Table E5-1 lists the CDF and LERF values from the baseline Model of Record (MOR) FPIE Revision 5.3 (including internal flooding) and Fire PRA Revision 5.3 models (References 9 and 10, respectively).

Table E5-1: Total Baseline Model of Record CDF/LERF

Prairie Island Unit 1 Baseline CDF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3 MOR)	1.28E-05
Fire PRA (Fire Rev 5.3 MOR)	6.64E-05
Seismic CDF[1]	3.00E-06
Other External Events	No significant contribution
Total Unit 1 CDF	**8.22E-05**

Prairie Island Unit 1 Baseline LERF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3 MOR)	2.15E-07
Fire PRA (Fire Rev 5.3 MOR)	9.64E-07
Seismic LERF[1]	1.50E-07
Other External Events	No significant contribution
Total Unit 1 LERF	**1.33E-06**

Prairie Island Unit 2 Baseline CDF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3 MOR)	1.25E-05
Fire PRA (Fire Rev 5.3 MOR)	6.61E-05
Seismic CDF[1]	3.00E-06
Other External Events	No significant contribution
Total Unit 2 CDF	**8.16E-05**

Prairie Island Unit 2 Baseline LERF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3 MOR)	1.86E-07
Fire PRA (Fire Rev 5.3 MOR)	9.27E-07
Seismic LERF[1]	1.50E-07
Other External Events	No significant contribution
Total Unit 2 LERF	**1.26E-06**

Table E5-1 Notes:

1. Based on the seismic CDF and LERF penalty factors calculated in Enclosure 4.

Table E5-2 lists the CDF and LERF values from the baseline application-specific PRA Models FPIE Rev 5.3-APP1 (including internal flooding) and fire PRA Rev 5.3-APP1 models (References 11 and 12, respectively).

Table E5-2: Total Baseline Application Specific Model CDF/LERF

Prairie Island Unit 1 Baseline CDF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3-APP1)	1.21E-05
Fire PRA (Fire Rev 5.3-APP1)	6.60E-05
Seismic CDF[1]	3.00E-06
Other External Events	No significant contribution
Total Unit 1 CDF	**8.11E-05**

Prairie Island Unit 1 Baseline LERF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3-APP1)	1.87E-07
Fire PRA (Fire Rev 5.3-APP1)	9.60E-07
Seismic LERF[1]	1.50E-07
Other External Events	No significant contribution
Total Unit 1 LERF	**1.30E-06**

Prairie Island Unit 2 Baseline CDF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3-APP1)	1.22E-05
Fire PRA (Fire Rev 5.3-APP1)	6.59E-05
Seismic CDF[1]	3.00E-06
Other External Events	No significant contribution
Total Unit 2 CDF	**7.52E-05**

Prairie Island Unit 2 Baseline LERF	
Source	**Contribution**
Internal Events PRA (FPIE Rev 5.3-APP1)	1.83E-07
Fire PRA (Fire Rev 5.3-APP1)	9.26E-07
Seismic LERF[1]	1.50E-07
Other External Events	No significant contribution
Total Unit 2 LERF	**1.26E-06**

Table E5-2 Notes:

1. Based on the seismic CDF and LERF penalty factors calculated in Enclosure 4.

As demonstrated in Tables E5-1 and E5-2, the total CDF and total LERF for both the PRA models of record and application-specific PRA models are within the guidelines set forth in RG 1.174 and support small changes in risk that may occur during RICT entries following implementation of the RICT Program. Therefore, the proposed PINGP RICT Program implementation is consistent with NEI 06-09-A guidance.

3.0 REFERENCES

1. Letter from the NRC to NEI, “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, ‘Risk-Informed Technical Specifications Initiative 4B,

Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)", dated May 17, 2007 (ADAMS Accession No. ML071200238)

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

3. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Revision 2, dated May 2011 (ADAMS Accession No. ML100910006)

4. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 2, dated March 2009 (ADAMS Accession No. ML090410014)

5. Letter (L-PI-18-012) from NSPM to the NRC, "Application to Adopt 10 CFR 50.69, 'Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors'", dated July 20, 2018 (ADAMS Accession No. ML18204A393)

6. Letter (L-PI-18-005) from NSPM to the NRC, "License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805", dated May 18, 2018 (ADAMS Accession No. ML18138A402)

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

8. Letter (L-PI-18-005) from NSPM to the NRC, "License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805", dated May 18, 2018 (ADAMS Accession No. ML18138A402)

9. NSPM PRA Document PRA-PI-QU, "PRA Level 1 Quantification", Revision 5.3, dated November 2017

10. NSPM PRA Document FPRA-PI-FQ, "Fire PRA Quantification Notebook", Revision 5.3, dated April 2018

11. NSPM PRA Document V.SMN.18.009, "Application Specific Model FPIE Rev 5.3-APP1 and BE Tag File Development", Revision 0, dated December 2018

12. NSPM PRA Document V.SMN.18.005, "FPRA Rev 5.3-APP1 Application Specific Model", Revision 0, dated June 2018

ENCLOSURE 6

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

JUSTIFICATION OF APPLICATION OF AT-POWER PRA MODELS TO SHUTDOWN MODES

(1 Page Follows)

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

1.0 INTRODUCTION

This enclosure is not applicable to the Prairie Island Nuclear Generating Plant submittal. Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy, is proposing to apply the Risk-Informed Completion Time Program only in Modes 1 and 2.

ENCLOSURE 7

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

PRA MODEL UPDATE PROCESS

(3 Pages Follow)

PRA Model Update Process

1.0 INTRODUCTION

Section 4.0, Item 8 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0 (Reference 2), requires that the license amendment request (LAR) provide a discussion of the licensee's programs and procedures which assure the PRA models supporting the RMTS are maintained consistent with the as-built/as-operated plant. Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), maintains a process and procedure to maintain and update the Probabilistic Risk Assessment (PRA) models in manner to ensure these models reflect the as-built, as-operated plant. For NSPM, a single PRA model of record (MOR) is used to evaluate plant risks for Prairie Island Nuclear Generating Plant (PINGP), Unit 1 and Unit 2.

This enclosure describes the administrative controls and procedural processes applicable to the configuration control and update of the Probabilistic Risk Assessment (PRA) models used to support the Risk-Informed Completion Time (RICT) Program, which will be in place to ensure that these models reflect the as-built/as-operated plant. Plant changes, including physical modifications and procedure revisions, will be identified and reviewed prior to implementation to determine if they could impact the PRA models per the PRA Change Database and Application Guide (Reference 3) and the PRA Model Maintenance and Update (Reference 4). The PRA model update process will ensure these plant changes are incorporated into the PRA models as appropriate. The process will include discovered conditions and errors associated with the PRA models which will be addressed in the applicable site Corrective Action Program (CAP).

Should a plant change or a discovered condition be identified with potential significant impact to the RICT Program calculations as defined by the plant procedures (Reference 3 and 4), an unscheduled update of the PRA model will be implemented. Otherwise, the PRA model change is incorporated into a subsequent periodic model update. Such pending changes are considered when evaluation other changes until there are fully implemented into PRA models. Periodic updates are nominally performed every two fuel cycles.

2.0 PRA MODEL UPDATE PROCESS

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

The NSPM fleet risk management process and model governance ensures that the applicable PRA MOR and application-specific models used for the RICT Program reflects the as-built, as-operated plant for both PINGP Unit 1 and Unit 2. The PRA model update process delineates the responsibilities and guidelines for controlling and updating the full power internal events, internal flood and fire PRA models including both the periodic and unscheduled PRA model updates.

The process includes provisions to track, evaluate and prioritize potential impact areas affecting the technical elements of the PRA models (e.g., due to plant changes, plant/industry operational experience, or errors or limitations identified in the model), assessing the individual and cumulative risk impact of unincorporated changes, and controlling the model and necessary computer files, including those associated with the Configuration Risk Management (CRM) model.

2.2 Review of Plant Changes for Incorporation into the PRA Model

1. The NSPM PRA Change Database (PCD) is the tool used to identify and track all PRA model changes including physical modifications to the facility and to operating practices and procedures with consideration of both temporary and permanent changes. Changes with potential significant risk impact are tracked using the NSPM PRA Change Database (PCD) and the CAP.

2. Plant changes or discovered conditions captured in the PCD are subject to an applicability review for potential impacts to the PRA models including the CRM model and the subsequent risk calculations which support the RICT Program (NEI 06-09-A, Section 2.3.4, Items 7.2 and 7.3, and Section 2.3.5, Items 9.2 and 9.3).

3. Plant changes are preliminary evaluated and screened based on risk criteria consistent with fleet procedural requirements (References 3 and 4) with consideration of the cumulative impact of other pending changes. Changes with potential for significant impact will be incorporated in an unscheduled update and application-specific PRA model(s), consistent with the NEI 06-09-A guidance (Section 2.3.5, Item 9.2) with the PRA model published the following quarter. These changes are also addressed in the CAP.

4. Otherwise, the change is assigned a priority and is incorporated at a subsequent periodic updated consistent with fleet procedural requirements (Reference 4).

5. PRA MOR updates for the PINGP unit(s) changes are nominally performed once every two fuel cycles, but may be sooner or later depending on plant needs and management discretion.

6. If a PRA model change is required for the CRM model, but cannot be immediately implemented for a significant plant change or discovered condition, one of the following is applied:

    a. Analysis to address the expected risk impact of the change via risk-informed screening criteria will be performed. In such a case, these analyses become part of the RICT Program calculation process until the plant changes are incorporated into the published PRA model and within the appropriate time associated with the priority of the update.

b. The application and use of such bounding analyses, as appropriate, may serve as quantitative analyses to support the expected risk impact of the change and is consistent with the guidance of NEI 06-09-A.

c. Appropriate administrative restrictions on the use of the RICT program for extended Completion Time are put in place until the model changes are completed, consistent with the guidance of NEI 06-09-A, Section 2.3.5, Item 9.3.

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

3.0 REFERENCES

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

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

3. NSPM Procedure FP-PE-PRA-01, “PRA Change Database Use and Application Guide”, Revision 9

4. NSPM Procedure FP-PE-PRA-02, “PRA Guideline for Model Maintenance and Update”, Revision 17

ENCLOSURE 8

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

ATTRIBUTES OF THE CONFIGURATION RISK MANAGEMENT MODEL

(3 Pages Follow)

Attributes of the Configuration Risk Management Model

1.0 INTRODUCTION

Section 4.0, Item 9 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines", Revision 0 (Reference 2), requires that the license amendment request (LAR) provide a description of the probabilistic risk assessment (PRA) models and tools used to support the RMTS. This includes identification of how the baseline probabilistic risk assessment (PRA) model will be modified for use in the Configuration Risk Management (CRM) tools, quality requirements applied to the PRA models and CRM tools, consistency of calculated results from the PRA and CRM model, and training and qualification programs applicable to personnel responsible for development and use of the CRM tools. This item should also confirm that the RICT Program tools can be readily applied for each Technical Specification (TS) Limiting Conditions for Operation (LCO) within the scope of the plant-specific submittal.

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

a) To preserve the core damage frequency (CDF) and large early release frequency (LERF) quantitative results;

b) To maintain the quality of the peer-reviewed PRA models; and

c) To correctly accommodate changes in risk due to configuration-specific consideration.

Quality control and training programs applicable to the RICT Program are also discussed in this enclosure.

2.0 TRANSLATION OF BASELINE MODEL FOR USE IN CONFIGURATION RISK

The baseline PRA model for internal events, including internal flood and internal fire, are peer-reviewed models. These models are updated when necessary to incorporate plant changes to reflect the as-built/as-operate plant as discussed in Enclosure 7. The internal flood model is integrated in the internal events model. The internal fire PRA model is maintained as a separate model. These PRA models will be used in the RICT Program. The models may be optimized for quantification speed, but will be verified to provide results equivalent to the baseline models and in accordance with approved procedures.

The CRM software will be used to facilitate all configuration-specific risk calculations and support RICT Program implementation. The baseline PRA models will be modified to create a single top model as follows:

- The unit availability factor is set to 1.0 (unit available).
- Maintenance unavailability is set to zero/false, unless unavailable due to the configuration.
- Mutually exclusive combinations, including normally disallowed maintenance combinations, are adjusted to allow accurate analysis of the configuration.
- For systems where some trains are in service and some in standby, the CRM model addresses the actual configuration of the plant including defining in service trains as needed.

The CRM software is designed to quantify the unit-specific configuration for both internal events, including internal flooding and fire, and includes the seismic risk contribution when calculating the risk management action time (RMAT) and RICT. The unique aspect of the CRM software for the RICT Program will be the quantification of the fire risk and the inclusion of the seismic risk contribution.

3.0 QUALITY REQUIREMENT AND CONSISTENCY OF PRA MODEL AND CONFIGURATION RISK MANAGEMENT TOOLS

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

The information provided in Enclosure 2 demonstrates that the PINGP internal events (including internal flooding) and internal fire PRA models reasonably conform to the associated industry standards endorsed by Regulatory Guide (RG) 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2 (Reference 3). This information provides a robust basis for concluding that the PRA models are of sufficient quality for use in risk-informed licensing actions.

For maintenance of an existing CRM model, changes made to the baseline PRA model in translation to the CRM model will be controlled and documented in accordance with Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), PRA procedures (Reference 4 and Reference 5). These procedures address the process for identification and corrective actions to evaluate and disposition model errors and changes to ensure models are accurate, as described in Enclosure 7. Acceptance testing is performed after every configuration risk model update to ensure that the software works as intended and that quantification results are reasonable. The CRM model is nominally updated to reflect the as-built, as-operated plant once every two fuel cycles, but may be sooner or later depending on plant needs and management discretion.

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

4.0 TRAINING AND QUALIFICATION

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

5.0 APPLICATION OF THE CONFIGURATION RISK TOOL TO THE RICT PROGRAM SCOPE

The Electric Power Research Institute (EPRI) Phoenix Risk Monitor software, or equivalent, will be used to facilitate all configuration-specific risk calculations and support the RICT Program implementation. This program is specifically designed to support the implementation of RMTS. The Phoenix Risk Monitor software will permit the user to evaluate all plant configurations using appropriate mapping of plant equipment to the PRA basic events. The equipment in the scope of the RICT Program shall be able to be evaluated in the appropriate PRA models. The Phoenix Risk Monitor software implementation will conform to NSPM software quality assurance requirements.

6.0 REFERENCES

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

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

3. NRC Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities”, Revision 2, dated March 2009 (ADAMS Accession No. ML090410014)

4. NSPM Procedure FP-PE-PRA-01, “PRA Change Database Use and Application Guide”, Revision 9

5. NSPM Procedure FP-PE-PRA-02, “PRA Guideline for Model Maintenance and Update”, Revision 17

ENCLOSURE 9

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

KEY ASSUMPTIONS AND SOURCES OF UNCERTAINTY

(5 Pages Follow)

Key Assumptions and Sources of Uncertainty

1.0 INTRODUCTION

The purpose of this enclosure is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for the Risk Informed Completion Time (RICT) Program. Nuclear Energy Institute (NEI) topical report NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 1), Section 2.3.4, item 10 requires an evaluation to determine insights that will be used to develop risk management actions (RMAs) to address these uncertainties. The baseline internal events PRA (including internal flood) and fire PRA models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to the RICT Program calculations, to perform sensitivity analyses where appropriate, to discuss the results and to provide dispositions for the RICT Program. The PINGP internal events (including internal flooding) and fire PRA models described within this LAR are the same as those described within Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), submittals regarding adoption of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (Reference 2), and modification of the list of required NFPA 805 modifications (Reference 3), respectively.

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

2.0 ASSESSMENT OF INTERNAL EVENTS PRA EPISTEMIC UNCERTAINTY IMPACTS

In order to identify key sources of uncertainty for RICT Program application, an evaluation of internal events baseline PRA model uncertainty was performed, based on the guidance in NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, Revision 1 (Reference 4) and Electric Power Research Institute (EPRI) Technical Report (TR)-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments” (Reference 5). As described in NUREG-1855, Revision 1, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the Prairie Island Nuclear Generating Plant (PINGP) baseline PRA model quantification (Reference 6). Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the PINGP internal events PRA technical elements are noted in the individual notebooks. The internal events PRA model uncertainties evaluation is documented in Reference 5 and considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. EPRI compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference 4), and the evaluation performed for PINGP (Reference 7) considered each of the generic sources of modeling uncertainty as well as the plant-specific sources. A specific evaluation of the impact of identified uncertainties for this LAR was performed in Reference 8.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA (Reference 7) and are then considered for their impact (Reference 8) on this LAR. No specific issues of PRA completeness have been identified relative to this LAR, based on the results of the internal events PRA (including internal flood) review.

Based on the review of sources (Reference 8) of uncertainty, no specific uncertainty issues have been identified that would impact the RICT application.

3.0 ASSESSMENT OF FIRE PRA EPISTEMIC UNCERTAINTY IMPACTS

The purpose of the following discussion is to address the epistemic uncertainty in the PINGP fire PRA. The fire PRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the fire PRA and because the state of knowledge in these elements continues to evolve. The development of the PINGP fire PRA was guided by NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology” (Reference 9), and the fire PRA model used consensus models described in NUREG/CR-6850. Enclosure 2 provides a detailed discussion of the peer review Facts and Observations (F&Os) and the resolutions.

The PINGP fire PRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Further, appropriate fire impacts were identified for the systems modeled in the internal events PRA and were addressed in the fire PRA. Fire PRA methods were based on NUREG/CR-6850, as well as other more recent NUREGs (e.g., NUREG-7150, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)” (Reference 11), NUREG-2169, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database” (Reference 12), and NUREG-2178, “Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE) – Volume 1: Peak Heat Release Rates and Effects of Obstructed Plume” (Reference 13), and published “frequently asked questions” (FAQs) for the fire PRA.

NSPM used guidance provided in NUREG-1855 and EPRI TR-1026511, “Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty” (Reference 14), to review plant-specific and generic uncertainties associated with the fire PRA for the RICT Program application. The potential sources of model uncertainty in the PINGP fire PRA model were evaluated for their potential impacts on the RICT calculations in Reference 8. The review identified no specific uncertainty issues that would impact the RICT application.

4.0 ASSESSMENT OF TRANSLATION (REAL TIME RISK MODEL) UNCERTAINTY IMPACTS

Incorporation of the baseline PRA models into the Real Time Risk model used for RICT Program calculations may introduce new sources of model uncertainty. Table E9-1 provides a description of the relevant model changes and dispositions of whether any of the changes made represent possible new sources of model uncertainty that must be addressed. Refer to Enclosure 8 for additional discussion on the Real Time Risk model.

Table E9-1: Assessment of Translation Uncertainty Impacts

Real Time Risk Model Change and Assumptions	Part of Model Affected	Impact on Model	Disposition
PRA model logic structure may be optimized to increase solution speed.	Fault tree logic model structure, affecting both internal events and fire PRAs	The model, if restructured, will be logically equivalent and produce results comparable to the baseline PRA logic model	Since the restructured model will produce comparable numerical results, this is not a source of uncertainty for the RICT Program.
Incorporation of seismic risk bias to support RICT Program risk calculations. A conservative value for the seismic delta CDF is applicable.	Calculation of RICT and risk management action time (RMAT) within Real Time Risk Model	The addition of bounding impacts for seismic events has no impact on baseline PRA or Real Time Risk Model. Impact is reflected in calculation of all RICTs and RMATs.	Since this is a bounding approach for addressing seismic risk in the RICT Program, it is not a source of translation uncertainty, and RICT Program calculations are not impacted. Therefore, no mandatory RMAs are required.
Set plant availability (Reactor Critical Years Factor) basic event to 1.0.	Initiating event frequency values in internal events and fire PRAs	Since the Real Time Risk model evaluates specific configurations during at-power conditions, the assumption of a plant availability factor that is less than 1.0 is not appropriate. Adjustment of the initiating event frequencies allows the Real Time Risk Model to	This change is consistent with Real Time Risk Tool practice; therefore, this change does not represent a source of translation uncertainty and RICT Program calculations are not impacted. Therefore, no mandatory RMAs are required.

Table E9-1: Assessment of Translation Uncertainty Impacts

Real Time Risk Model Change and Assumptions	Part of Model Affected	Impact on Model	Disposition
		produce appropriate results for specific at-power configurations.	

5.0 REFERENCES

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

2. Letter (L-PI-18-012) from NSPM to the NRC, “Application to Adopt 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors’”, dated July 20, 2018 (ADAMS Accession No. ML18204A393)

3. Letter (L-PI-18-005) from NSPM to the NRC, “License Amendment Request to Revise License Condition Associated with Implementation of NFPA 805”, dated May 18, 2018 (ADAMS Accession No. ML18138A402)

4. NRC NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466)

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

6. NSPM PRA Document PRA-PI-QU, “PRA Level 1 Quantification”, Revision 5.3, dated November 2017

7. NSPM PRA Document FPRA-PI-FQ, “Fire PRA Quantification Notebook”, Revision 5.3, dated April 2018

8. NSPM Calculation V.SPA.19.013, “Prairie Island RICT Evaluation of Open F&Os and Uncertainties”, Revision 0, dated November 21, 2019

9. NRC NUREG/CR-6850 Volume 2, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology”, dated September 2005 (ADAMS Accession No. ML15167A411)

10. NRC NUREG/CR-6850, Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancements”, dated September 2010 (ADAMS Accession No. ML103090242)

11. NRC NUREG/CR 7150, Volume 1, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure”, dated October 2012 (ADAMS Accession No. ML12313A105)

12. NRC NUREG-2169, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009”, Revision 0, dated January 2015 (ADAMS Accession No. ML15016A069)

13. NRC NUREG-2178, Volume 1, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE) – Volume 1: Peak Heat Release Rates and Effects of Obstructed Plume”, Revision 0, dated April 2016 (ADAMS Accession No. ML16110A140)

14. EPRI Technical Report TR-1026511, “Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty”, dated December 2012

ENCLOSURE 10

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

PROGRAM IMPLEMENTATION

(4 Pages Follow)

Program Implementation

1.0 INTRODUCTION

Section 4.0, Item 10 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 2), requires that the license amendment request (LAR) provide a description of the implementing programs and procedures regarding plant staff responsibilities for the Risk Managed Technical Specifications (RMTS) Implementation, and specifically discuss the decision process for risk management action (RMA) implementation during a Risk-Informed Completion Time (RICT). Several procedures and processes are detailed in other enclosures that are not repeated in this enclosure addressing Probabilistic Risk Assessment (PRA) Model Update, Cumulative Risk Assessment, Monitoring Program and Risk Management Actions.

This enclosure provides a description of the implementing programs and the administrative controls and procedures regarding the plant staff responsibilities for the RICT Program, including training of plant personnel, and specifically discusses the decision process for RMA implementation during extended Completion Times (CT).

2.0 RICT PROGRAM AND PROCEDURES

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter “NSPM”), will develop a program description and implementing procedures for the RICT Program. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT Program. More detailed procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT Program. The program description and implementing procedures will incorporate the programmatic requirements for the RMTS included in NEI 06-09-A. The program will be integrated with the online work control process. The work control process currently identifies the need to enter a Limiting Conditions for Operation (LCO) Action statement as part of the planning process and will additionally identify whether the provisions of the RICT Program are requirements for the planned work. The risk thresholds associated with 10 CFR 50.65(a)(4) performance monitoring provisions and Mitigating System Performance Index (MSPI) thresholds will assist in controlling the amount of risk expended in use of the RICT Program (Reference 1, Table 3-1).

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

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

- Plant management positions with authority to approve entry into RICT Program.
- Important definitions related to the RICT Program.
- Departmental and position responsibilities for activities in the RICT Program.
- Plant conditions for which the RICTs under voluntary and emergent conditions.
- Limitations on implementing RICTs under voluntary and emergent conditions.
- Implementation of the RICT and risk management action time (RMAT) within 12 hours or within the most limiting front-stop CT after a plant configuration change.
- Requirement to identify and implement RMAs when the RMAT is exceeded or is anticipated to be exceeded, and to consider common cause failure potential in emergent RICTs.
- Guidance on the use of RMAs including the conditions under which they may be credited in RICT calculations.
- Conditions for exiting a RICT.
- Documentation requirements related to individual RICT evaluations, implementation of extended CTs, and accumulated annual risk.

3.0 RICT PROGRAM TRAINING

The scope of training for the RICT Program will include rules for the new TS program, Configuration Risk Management (CRM) software (Electric Power Research Institute (EPRI) Phoenix Risk Monitor), TS Actions included in the program, and procedures. This training will be conducted for the following NSPM personnel:

<u>Site Personnel</u>

- Operations Manager
- Operations Personnel (Licensed and Non-Licensed)
- Outage Manager
- Plant Manager
- Work Planning Personnel
- Work Week Managers
- Regulatory Affairs Personnel
- Selected Maintenance Personnel
- Other Selected Management

Fleet Support & Corporate Personnel

- Operations Corporate Functional Area Manager
- Operations Training
- Regulatory Affairs Personnel
- Risk Management Personnel and Managers
- Training Management and Personnel
- Engineering
- Other Selected Management

Training will be carried out in accordance with the NSPM training procedures and processes. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation requirements, as developed and maintained by the Nation Academy for Nuclear Training. NSPM has planned two levels of training for the implementation of the RICT Program. They are described below:

3.1 Level 1 Training

This the most detailed training. It is intended for the individual who will be directly involved in the implementation of the RICT Program. This level of training includes the following attributes:

- Specific training on the revised TS
- Record keeping requirements
- Case studies
- Hands-on experience with the CRM tool for calculating RMAT and RICT.
- Identifying appropriate RMAs
- Common cause failure RMA considerations in emergent RICTs
- Other detailed aspects of the RICT Program

3.2 Level 2 Training

This training is applicable to plant management positions with authority to approve entry into the RICT Program, as well as supervisors, managers, and other personnel who will closely support RICT implementation. Additionally, this training with be given to remaining personnel who require an awareness of the RICT Program. These individuals need a broad understanding of the purpose, concepts, and limitations of the RICT Program. Level 2 training is different from Level 1 training in that hands-on time with the Real Time Risk Tool, case studies, and other specifics are not required.

All of the above training will be conducted within the procedural guidance set forth in NSPM's training and qualification procedures, unless otherwise noted.

4.0 REFERENCES

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

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

ENCLOSURE 11

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

MONITORING PROGRAM

(2 Pages Follow)

Monitoring Program

1.0 INTRODUCTION

Section 4.0, Item 12 of the NRC Final Safety Evaluation (SE) (Reference 1) for NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 2), requires that the license amendment request (LAR) provide a description of the implementing and monitoring program as described in Regulatory Guide (RG) 1.174, “An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”, Revision 1 (Reference 3), and NEI 06-09-A. (Note that Revision 2 of RG 1.174 (Reference 4) was issued by the NRC in May 2011 which made editorial changes to the applicable section referenced in the NRC SE for Section 4.0, Item 12.)

This enclosure provides a description of the process applied to govern and monitor calculation of cumulative risk impact in support of the Risk-Informed Completion Time (RICT) Program, specifically the calculation of cumulative risk of extended Completion Times (CTs). Calculation of the cumulative risk for the RICT Program is discussed in Step 14 of Section 2.3.1 and Step 7.1 of Section 2.3.2 of NEI 06-09-A. General requirements for a Performance Monitoring Program for risk-informed applications are discussed in Element 3 of the RG 1.174, Revision 2.

2.0 DESCRIPTION OF MONITORING PROGRAM

The RICT Program will require calculation of cumulative risk impacts at least once every 2 fuel cycles. For the assessment period under evaluation, plant and system historical data is collected to establish the risk increase associated with each application of an extended CT for both core damage frequency (CDF) and large early release frequency (LERF). The total risk impact will be calculated by summing all risk associated with each RICT application. This summation is the change in CDF or LERF above the zero maintenance baseline levels during the period of operation in the extended CT (i.e., beyond the front-stop CT). The change in risk will be converted to average annual values and documented every two fuel cycles.

The total average annual change in risk for extended CTs will be compared to the guidance of RG 1.174, Revision 2, Figures 4 and 5, acceptance guidelines for CDF and LERF, respectively. If the actual annual risk increase is acceptable (i.e., not in Region I of Figures 4 and 5 of RG 1.174, Revision 2), then RICT Program implementation is acceptable for the assessment period. Otherwise, further assessment of the cause of exceeding the acceptance guidelines of RG 1.174, Revision 2, and implementation of any necessary corrective actions to ensure future plant operation is within the guidelines will be conducted under the corrective action program (CAP).

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

- RICT applications that dominated the risk increase

- Risk contributions from planned vs. emergent RICT applications
- Risk Management Actions (RMA) implemented but not credited in the risk calculations
- Risk impact from applying RICT to avoid multiple shorter duration outages

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

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

In addition to impacting cumulative risk, the implementation of the RICT Program may potentially impact the unavailability of SCCs. The Maintenance Rule (MR) monitoring programs under 10 CFR 50.65 provide for evaluation and disposition of unavailability impacts which may be incurred from implementation of the RICT Program. The SSCs in the scope of the RICT Program which are also in the scope of the MR allows the use of the MR Program.

The monitoring program of the MR, along with the specific assessment of cumulative risk impact described above, serve as the Implementation and Monitoring Program for the RICT Program as described in Element 3 of RG 1.174, Revision 1, and NEI 06-09-A.

3.0 REFERENCES

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

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

3. NRC Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”, Revision 1, dated November 2002 (ADAMS Accession No. ML023240437)

4. NRC Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis”, Revision 2, dated May 2011 (ADAMS Accession No. ML100910006)

ENCLOSURE 12

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

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

RISK MANAGEMENT ACTION EXAMPLES

(9 Pages Follow)

Risk Management Action Examples

1.0 INTRODUCTION

This enclosure describes the process for identification and implementation of Risk Management Actions (RMA) applicable during extended Completion Times (CT) and provides examples of RMAs. RMAs will be governed by plant procedures for planning and scheduling maintenance activities. The procedures will provide guidance for the determination and implementation of RMAs when entering the Risk-Informed Completion Time (RICT) Program consistent with the guidance provided in Nuclear Energy Institute (NEI) 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 1).

2.0 RESPONSIBILITIES

For planned entries into the RICT Program, the Work Management Department is responsible for developing the RMAs with assistance from the Operations and Risk Management Departments. Operations is responsible for approval and implementation of RMAs. For emergent entry into extended CTs, Operations is also responsible for developing the RMAs.

3.0 PROCEDURAL GUIDANCE

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

For emergent entry into a RICT, if the extent of condition is not known, RMAs related to the success of redundant and diverse SSCs and reducing the likelihood of initiating events relying on the affected function will be developed and implemented to address the increased likelihood of a common cause event.

RMAs will be implemented in accordance with current procedures (e.g., References 2, 3, and 4) no later than the time at which an incremental core damage probability (ICDP) of 1E-6 is reached, or no later than the time when an incremental large early release probability (ILERP) of 1E-7 is reached. If, as the result of an emergent condition, the instantaneous core damage frequency (ICDF) or the instantaneous large early release frequency (ILERF) exceeds 1E-3 per year or 1E-4 per year, respectively, RMAs are also required to be implemented. These requirements are consistent with the guidelines of NEI 06-09-A.

By determining which structures, systems, or components (SSCs) are most important from a CDF or LERF perspective for a specific plant configuration, RMAs may be created to protect these SSCs. Similarly, knowledge of the initiating event or sequence contribution to the configuration-specific CDF or LERF allows development of RMAs that enhance the capability to mitigate such events. The guidance in NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making” (Reference 5), and EPRI TR-1026511, “Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty” (Reference 6), will be used in examining PRA results for significant contributors for the configuration, to aid in identifying appropriate compensatory measures (e.g., related to risk-significant systems that may provide diverse protection or important support systems).

If the planned activity or emergent condition includes an SSC that is identified to impact fire PRA, as identified in the current Real Time Risk Program, fire PRA specific RMAs associated with that SSC will be implemented per the current plant procedure.

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

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

1) Actions to increase risk awareness and control.

   - Shift brief
   - Pre-job brief
   - Training
   - Presence of strategic engineer or other expertise related to the activity
   - Special purpose procedure to identify risk sources and contingency plans

2) Actions to reduce the duration of maintenance activities.

   - Pre-staging materials
   - Conducting training on mock-ups
   - Performing the activity around the clock
   - Performing walk-downs on the actual system(s) to be worked on prior to beginning work

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

   - Suspend or minimize activities on redundant systems
   - Suspend or minimize activities on other systems that adversely affect the CDF or LERF
   - Suspend or minimize activities on systems that may cause a trip or transient to minimize the likelihood of an initiating event that the out-of-service component is meant to mitigate
   - Use temporary equipment to provide backup power, ventilation, etc.

- Reschedule other risk-significant activities

4.0 EXAMPLES

Multiple example RMAs that may be considered during a RICT Program entry to reduce the risk impact and ensure adequate defense-in-depth are provided below. Specific examples are given for unavailability of one Diesel Generator (DG), one Offsite Source, one Battery Charger, or one Residual Heat Removal (RHR) pump.

A. Diesel Generator (Using the D1 DG as an example):

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established
  - Specific focus areas would be to review appropriate emergency or abnormal operating procedures for:
    - Loss of Offsite Power (LOOP) or Station Blackout events
      - Cross-tie actions to opposite unit
    - Loss of Secondary Heat Sink events
    - Component Cooling Malfunction events

- Perform a walkdown and validation of D2 to validate standby/readiness condition

- Perform a walkdown and validation of the 11 and 12 Auxiliary Feedwater (AFW) trains to validate standby/readiness condition

- Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for the following Fire Compartments:
  - 18: Relay and Cable Spreading Room, Unit 1 and 2
  - 58: Auxiliary Building Ground Floor

- For the above fire compartments, minimize the accumulation of transient combustibles in accordance with the station Fire Protection Program

- Notification of the transmission system operator (TSO) of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.
  - Discuss projected grid loading conditions with the TSO to identify if a planned entry into DG unavailability should be deferred

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.

- Confirmation of parts availability prior to entry into a preplanned RICT.

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

- Proactively implement RMAs during times of high grid stress conditions, such as during high demand conditions.
- Evaluate weather conditions for threats to the reliability of offsite power supplies.
- Defer elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with both units.
- Defer planned maintenance or testing that affects the reliability of OPERABLE DGs (D2, D5 and D6) and their associated support equipment which affect common system availability. Treat these as protected equipment.
- Maintain other unit DGs (D5 and D6) and buses (Bus 25 and Bus 26) available to allow crosstie from other unit to energize Engineered Safety Feature (ESF) buses.
- Defer planned maintenance or testing on redundant train safety systems. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.
- Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected DGs, as required.
- Implement equipment protection schemes in accordance with NSPM procedure FP-OP-PEQ-01 (Reference 7), as required.
- Maintain detection, suppression, and fire zone barriers intact and minimize transient combustibles for those Fire Areas/ Zones identified as being significant for the configuration.

B. One offsite power source inoperable (1R Transformer)

1) Actions to increase risk awareness and control.
    - Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established
        - Specific focus areas would be to review appropriate emergency or abnormal operating procedures for:
            - LOOP or Station Blackout events
                - Cross-tie actions to opposite unit
            - Loss of Secondary Heat Sink events
            - Loss of coolant accident (LOCA) events
            - Steam generator tube rupture (SGTR) events

- Perform a walkdown and validation of the DGs to validate standby/readiness condition

- Perform a walkdown and validation of the 11 and 12 AFW trains to validate standby/readiness condition

- Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for the following Fire Compartments:
  - 58: Auxiliary Building Ground Floor
  - 59: Auxiliary Building Mezzanine Level

- For the above fire compartments, minimize the accumulation of transient combustibles in accordance with the station Fire Protection Program

- Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.
  - Discuss projected grid loading conditions with the TSO to identify if a planned entry into the transformer unavailability should be deferred

2) Actions to reduce the duration of maintenance activities.

   - For preplanned RICT entry, creation of a sub-schedule related to the specific evolution which is reviewed for personnel resource availability.

   - Confirmation of parts availability prior to entry into a preplanned RICT.

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

- Proactively implement RMAs during times of high grid stress conditions, such as during high demand conditions.

- Evaluate weather conditions for threats to the reliability of offsite power supplies.

- Defer elective maintenance in the switchyard, on the station electrical distribution systems, and on the other transformers associated with both units.

- Defer planned maintenance or testing that affects the reliability of OPERABLE DGs (D1, D2, D5 and D6) and their associated support equipment which affect common system availability. Treat these as protected equipment.

- Maintain opposite unit DGs and 4 kV safeguards buses available to allow crosstie from other unit.

- Defer planned maintenance or testing on safety systems. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.

- Implement 10 CFR 50.65(a)(4) fire-specific RMAs, as required.

- Implement equipment protection schemes in accordance with NSPM procedure FP-OP-PEQ-01, as required.

- Maintain detection, suppression, and fire zone barriers intact and minimize transient combustibles for those Fire Areas / Zones identified as being significant for the configuration.

C. One battery charger inoperable (using battery charger 11 BATT CHG as an example)

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established
    - Specific focus areas would be to review appropriate emergency or abnormal operating procedures for:
        - Loss of DC Power
            - Installation of portable battery charger
        - Loss of Secondary Heat Sink events

- Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for the following Fire Zones:
    - 20: Unit 1 4.16 kV Safeguards Switchgear (Bus 16)
    - 31: A Train Hot Shutdown Panel & Air Compressor/AFW Room

- For the above fire zones, minimize the accumulation of transient combustibles in accordance with the station Fire Protection program

- Notification of the TSO of the configuration so that any planned activities with the potential to cause a grid disturbance are deferred.
    - Discuss projected grid loading conditions with the TSO to identify if a planned entry into battery charger unavailability should be deferred

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.

- Confirmation of parts availability prior to entry into a preplanned RICT.

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

- Proactively implement RMAs during times of high grid stress conditions, such as during high demand conditions.
- Evaluate weather conditions for threats to the reliability of offsite power supplies.
- Defer elective maintenance in the switchyard, on the station electrical distribution systems, and on the main and auxiliary transformers associated with both units.
- Defer planned maintenance or testing that affects the reliability of OPERABLE DGs (D1, D2, D5 and D6) and their associated support equipment which affect common system availability. Treat these as protected equipment.
- Protection of the remaining DC electrical buses in that unit. Protect opposite unit power supplies for remaining pumps in loop affected by the inoperable SSC.
- Defer planned maintenance or testing on redundant train safety systems. If testing or maintenance activities must be performed, a review of the potential risk impact will be performed.
- Remove nonessential loads from battery to extend time voltage will remain above minimum required level.
- Pre-stage (move) Portable Battery Charger in room of unavailable Battery Charger
- Implement 10 CFR 50.65(a)(4) fire-specific RMAs, as required.

D. Residual Heat Removal (RHR) Pump (Using the 11 RHR pump as an example):

1) Actions to increase risk awareness and control.

- Brief the on-shift operations crew concerning the unit activities, including any compensatory measures established
    - Specific focus areas would be to review appropriate emergency or abnormal operating procedures for:
        - LOCA events
            - Implementation of Sump B recirculation
        - Component Cooling Malfunction events
- Perform a walkdown and validation of the 12 Emergency Core Cooling System (ECCS) train to validate standby/readiness condition

- Perform a walkdown and validation of the 11 and 12 AFW trains to validate standby/readiness condition

- Perform a walkdown and validation of the containment sump recirculation valves to validate standby/readiness condition

- Perform a walkdown of and confirm availability of applicable suppression, detection and fire barriers for the following Fire Zones:
    - 20: Unit 1 4.16 kV Safeguards Switchgear (Bus 16)
    - 31: A Train Hot Shutdown Panel and Air Compressor / AFW Room

- For the above fire zones, minimize the accumulation of transient combustibles in accordance with the station Fire Protection program

2) Actions to reduce the duration of maintenance activities.

- For preplanned RICT entry, creation of a sub schedule related to the specific evolution which is reviewed for personnel resource availability.

- Confirmation of parts availability prior to entry into a preplanned RICT.

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

1. Defer planned maintenance or testing that affects the RHR 1 B Pump and its associated support equipment and treat those SSCs as protected equipment.

2. Defer planned maintenance or testing that affects the Component Cooling Heat Exchanger (CC HX) or its associated support equipment and treat those SSCs as protected equipment.

3. Implement 10 CFR 50.65(a)(4) fire-specific RMAs associated with the affected RHR Pump.

4. Implement 10 CFR 50.65(a)(4) equipment protection schemes in accordance with NSPM procedure FP-OP-PEQ-01, as required.

5. Maintain detection, suppression, and fire zone barriers intact and minimize transient combustibles for those Fire Areas/Zones identified as being significant for the configuration.

5.0 REFERENCES

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

2. NSPM Fleet Procedure FP-OP-RSK-01, “Risk Monitoring and Risk Management”, Revision 10

3. NSPM Fleet Guidance Document FG-OP-RSK-01, “Configuration Risk Monitor User Guide”, Revision 3

4. NSPM Fleet Procedure FP-WM-IRM-01, “Integrated Risk Management”, Revision 19

5. NRC NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, Revision 1, dated March 2017 (ADAMS Accession No. ML17062A466)

6. EPRI Technical Report TR-1026511, “Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty”, dated December 2012

7. NSPM Fleet Procedure FP-OP-PEQ-01, “Protected Equipment Program”, Revision 23