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April 19, 2021

L-PI-21-007
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

<u>Application to Revise Technical Specifications to Adopt TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes"</u>

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

The proposed amendment will adopt TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes," with site-specific variations. This proposed change eliminates the defined term CORE ALTERATIONS and all uses of the term from the TS and Bases. The proposed amendment also adopts TSTF-571-T, "Revise Actions for Inoperable Source Range Neutron Flux Monitor," which changes TS 3.9.3, "Nuclear Instrumentation," to ensure that no actions are taken that could alter the core reactivity when a required core subcritical neutron flux monitor is inoperable. The proposed amendment also makes an administrative change to reformat the page numbering of TS Section 5.0 and remove unused pages for ease of future changes.

The Enclosure provides a description and assessment of the proposed changes. Attachment 1 to the Enclosure provides the existing TS pages marked up to show the proposed changes. Attachment 2 to the Enclosure provides revised (clean) TS pages. Attachment 3 to the Enclosure provides existing TS Bases pages marked to show the proposed changes for information only.

Approval of the proposed amendment is requested twelve months after acceptance. Once approved, the amendment shall be implemented within 90 days.

In accordance with 10 CFR 50.91(b)(1), a copy of this application, with the enclosure, is being provided to the designated Minnesota Official.

Please contact Mr. Jeff Kivi at (612) 330-5788 or Jeffrey.L.Kivi@xcelenergy.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 April 19, 2021.

Christopher P. Domingos
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

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

ENCLOSURE

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

License Amendment Request

TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes"

ATTACHMENTS

1. Proposed Technical Specification Changes (Mark-Up)
2. Revised Technical Specification Pages
3. Proposed Technical Specification Bases Changes (Mark-Up)

License Amendment Request

TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes"

1.0 SUMMARY DESCRIPTION

The proposed amendment will adopt TSTF-471-A, Revision 1, “Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes,” with site-specific variations. This proposed change eliminates the defined term CORE ALTERATIONS and all uses of the term from the TS and Bases. The proposed amendment also adopts TSTF-571-T, Revision 0, “Revise Actions for Inoperable Source Range Neutron Flux Monitor,” which changes TS 3.9.3, “Nuclear Instrumentation,” to ensure that no actions are taken that could alter the core reactivity when a required core subcritical neutron flux monitor is inoperable. The proposed amendment also makes an administrative change to reformat the page numbering of TS Section 5.0 and remove unused pages for ease of future changes.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

The OPERABILITY of the minimum AC sources, DC sources, inverters, and distribution subsystems during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;
b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The limit on the boron concentrations of the Reactor Coolant System (RCS) and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6.

Core subcritical neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed core subcritical neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors (N-31, N-32, N-51, and N-52) are located external to the reactor vessel and detect neutrons leaking from the core. The detectors provide continuous visual indication in the control room. The installed BF3 neutron flux monitors provide an audible indication to alert operators in containment to a possible dilution accident.

2.2 Current Technical Specifications Requirements

The following PINGP Units 1 and 2 TS include discussion of CORE ALTERATIONS:

- 1.1 Definitions includes a definition of CORE ALTERATIONS.

- TS 3.8.2, AC Sources – Shutdown, Required Actions A.1 and B.1 require suspension of CORE ALTERATIONS if a required path or a required diesel generator, respectively, is inoperable while in MODES 5 or 6 or during movement of irradiated fuel assemblies.
- TS 3.8.5, DC Sources – Shutdown, Required Action B.1 requires suspension of CORE ALTERATIONS if one required DC electrical power subsystem inoperable for reasons other than an inoperable battery charger while in MODES 5 or 6 or during movement of irradiated fuel assemblies.
- TS 3.8.8, Inverters – Shutdown, Required Action A.1 requires suspension of CORE ALTERATIONS if one required inverter is inoperable while in MODES 5 or 6 or during movement of irradiated fuel assemblies.
- TS 3.8.10, Distribution Systems – Shutdown, Required Action A.2.1 requires suspension of CORE ALTERATIONS if one or more required safeguards AC, DC, or Reactor Protection Instrument AC electrical power distribution subsystem is inoperable while in MODES 5 or 6 or during movement of irradiated fuel assemblies.
- TS 3.9.1, Boron Concentration, Required Action A.1 requires suspension of CORE ALTERATIONS if Reactor Coolant System (RCS) boron concentration is not within limits specified in the Core Operating Limits Report (COLR) while in MODE 6.
- TS 3.9.3, Nuclear Instrumentation, Required Actions A.1 and C.2 require suspension of CORE ALTERATIONS if one required core subcritical neutron flux monitor or a required core subcritical neutron flux monitor audible count rate circuit, respectively, is inoperable in MODE 6. Required Action A.2 requires suspension of operations that would cause introduction into the reactor coolant system (RCS), coolant with boron concentration less than required to meet the boron concentration of TS 3.9.1.

PINGP Units 1 and 2 TS, Section 5.0, contains administrative controls. All of TS, Section 5.0, is within the scope of this amendment request.

2.3 Reason for the Proposed Changes

Adopt TSTF-471

Suspending CORE ALTERATIONS has no effect on the initial conditions or mitigation of any Design Basis Accident (DBA) or transient, and these requirements apply an operational burden with no corresponding safety benefit.

Adopt TSTF-571-T

TSTF-571-T addresses NRC staff concerns with TSTF-51, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations,", TSTF-286, Revision 2, "Define 'Operations Involving Positive Reactivity Additions'," and TSTF-471.

Section 5.0 Page Numbering

The proposed change revises the pagination scheme used in Section 5.0 from the current scheme where page numbers continue from subsection to subsection to a new scheme where the page numbers reset with each subsection. Unused pages will also be removed. This administrative change will ease the implementation of future changes.

2.4 Description of the Proposed Changes

TSTF-471

The proposed change eliminates the defined term CORE ALTERATIONS from PINGP TS and all uses of the term from the Specifications and Bases. This affects the following Specifications:

1.1, Definitions, definition of CORE ALTERATIONS eliminated
3.8.2, A.C. Sources – Shutdown, Require Actions A.1 and B.1 eliminated
3.8.5, D.C. Sources – Shutdown, Required Action B.1 eliminated
3.8.8, Inverters – Shutdown, Required Action A.1 eliminated
3.8.10, Distribution Systems – Shutdown, Required Action A.2.1 eliminated
3.9.1, Boron Concentration, Required Action A.1 eliminated
3.9.3, Nuclear Instrumentation, Required Action A.1 modified to replace CORE ALTERATIONS with “positive reactivity additions”; Required Action C.2 eliminated.

TSTF-571-T

The proposed change revises TS 3.9.3, Required Action A.2 to add a note and revise the wording of the Required Action. The note provides an allowance to move fuel assemblies, sources, and reactivity control components if necessary to restore the inoperable core subcritical neutron flux monitor or to complete movement of a component to a safe condition. The Required Action is modified to suspend movement of fuel, sources, and reactivity control components within the reactor vessel.

Section 5.0 Page Numbering

The proposed change revises the Section 5.0 pagination scheme from the current scheme where pages numbers continue from subsection to subsection to a new scheme where the page numbers reset with each subsection. Unused pages will also be removed.

The Bases are revised as applicable to reflect the changes made. Attachment 1 to this Enclosure provides markups of the current TS. Attachment 3 of this Enclosure provides markups of the current TS Bases (for information).

3.0 TECHNICAL EVALUATION

3.1 Technical Evaluation of TSTF-471

The term “core alteration” does not appear in Title 10 of the Code of Federal Regulations. Since CORE ALTERATIONS only occur when the reactor vessel head is removed, it only applies in MODE 6. There are only two accidents considered during MODE 6 for PWRs: a fuel handling accident and a boron dilution accident. A fuel handling accident is initiated by the dropping of an irradiated fuel assembly, either in the containment or in the spent fuel pool. There are no mitigation actions. Suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of a fuel handling accident.

The second analyzed event is a boron dilution accident. A boron dilution accident is initiated by a dilution source which results in the boron concentration dropping below that required to maintain the SHUTDOWN MARGIN. As described in the Bases of Specification 3.9.1, "Boron Concentration," (which applies in MODE 6), "The refueling boron concentration and associated shutdown margin limits are specified in the COLR. The required boron concentration will vary depending on time in core life. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of keff ≤ 0.95 during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures." The accident is mitigated by stopping the dilution. Suspension of CORE ALTERATIONS has no effect on the mitigation of a boron dilution accident. Movement of control rods or fuel do not affect the initial conditions of a boron dilution accident as it is assumed that the control rods and fuel are in the most adverse conditions with a large safety margin (keff ≤ 0.95). To address the possibility of a misloaded fuel assembly in Technical Specification 3.9.3, a Required Action is added that suspends positive reactivity additions if nuclear instrumentation is not available. This precludes movement of fuel assemblies which could add reactivity to the core.

In summary, with the exception of suspending movement of irradiated fuel assemblies, there are no DBAs or transients that are initiated by or mitigation that is affected by suspension of CORE ALTERATIONS. Therefore, if all Required Actions that require suspension of CORE ALTERATIONS also require suspension of movement of irradiated fuel, suspension of CORE ALTERATIONS provides no safety benefit.

The term CORE ALTERATION is used in the following Specifications:

3.8.2, A.C. Sources - Shutdown
3.8.5, D.C. Sources - Shutdown
3.8.8, Inverters - Shutdown
3.8.10, Distribution Systems - Shutdown
3.9.1, Boron Concentration
3.9.3, Nuclear Instrumentation

An analysis of elimination of the defined term CORE ALTERATIONS for each of these Specifications is given below.

TS 3.8.2, TS 3.8.5, TS 3.8.8, and TS 3.8.10

If a required train of electrical power (A.C, D.C, Inverters, or Distribution Systems) is inoperable, the Required Actions require:

- Suspension of CORE ALTERATIONS
- Suspension of movement of irradiated fuel assemblies, and
- Suspension of operations involving positive reactivity additions that could result in loss of required shutdown margin (SDM) or boron concentration.

As discussed above, because the Required Actions require the suspension of movement of irradiated fuel assemblies, the initiating conditions for a fuel handling accident are prohibited. Because the Required Actions require the suspension of positive reactivity additions that could result in a loss of SDM, the initial conditions for a boron dilution accident are prevented. Therefore, the action to suspend CORE ALTERATIONS provides no safety benefit and is not needed.

<u>TS 3.9.1</u>

If boron concentration is not within limit, the Required Actions require immediate suspension of CORE ALTERATIONS, immediate suspension of positive reactivity additions, and immediate actions to restore the boron concentration within limits. This Specification is only concerned with a boron dilution accident, as SDM has no bearing on a fuel handling accident. Note that the Required Action to suspend positive reactivity additions would also prohibit adding fuel assemblies to the reactor, which could reduce the SDM. The requirement to suspend positive reactivity additions is the only action needed to mitigate a boron dilution event. Therefore, the action to suspend CORE ALTERATIONS provides no safety benefit and is not needed.

<u>TS 3.9.3, Required Actions A.1 and C.2</u>

If a required core subcritical neutron flux monitor (source range detector) is inoperable, CORE ALTERATIONS must be suspended and operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1. An inoperable source range detector has no effect on the initiation or mitigation of a fuel handling accident. Suspending introduction of RCS coolant at less than the required boron concentration prevents a boron dilution accident. Therefore, suspension of CORE ALTERATIONS does nothing to prevent any analyzed accident. It is a good practice to not move fuel with less than a full complement of source range detectors. Therefore, Required Action A.1 is revised to preclude positive reactivity additions. However, as stated in 10 CFR 50.36(b), "The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34." Suspension of CORE ALTERATIONS in this circumstance is not based on the analyses in the safety analysis report and that Required Action is removed.

In addition, PINGP TS 3.9.3 includes Required Action C.2 to suspend CORE ALTERATIONS when the required core subcritical neutron flux monitor audible count rate circuit is inoperable. Required Action C.2 is not part of standard TS and is proposed to be deleted consistent with the changes included in adopting TSTF-471 as well as NUREG-1431. This is acceptable because Condition C is included in PINGP TS to mitigate a postulated dilution accident. Suspending CORE ALTERATIONS has no effect on the initial conditions or mitigation of a dilution accident. The LCO discussion in the PINGP Bases for TS 3.9.3 states, "This LCO also requires that one audible count rate circuit, associated with either N-31 or N-32, be OPERABLE to ensure that audible indication is available to alert the operator in containment in the event of a dilution accident or improperly loaded fuel assembly.” However, the inclusion of a postulated improperly loaded fuel assembly in the discussion of the audible count rate circuit in this section is not consistent with the other sections of the PINGP TS 3.9.3 bases and is inconsistent with standard TS 3.9.3 bases, as well. Therefore, the action to suspend CORE ALTERATIONS provides no safety benefit and is not needed.

3.2 Technical Evaluation of TSTF-571-T

The proposed change revises TS 3.9.3, Required Action A.2 with a note to add allowance to move fuel assemblies, sources, and reactivity control components if necessary to restore operability or to complete movement of a component to a safe condition. The Required Action is further modified to suspend movement of fuel, sources, and reactivity control components within the reactor vessel. The proposed changes to Required Action A.2 protect against inadvertent boron dilution resulting in a change in core reactivity that the remaining core subcritical neutron flux monitor could not detect.

NRC letter to the Technical Specification Task Force (TSTF) dated November 7, 2013, (Reference 4) described NRC staff concerns with the following TSTF travelers:

- TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,”
- TSTF-286, Revision 2, “Operations Involving Positive Reactivity Additions,” and
- TSTF-471, Revision 1, “Eliminate Use of Term Core Alterations in Actions and Notes,"

NRC letter to the TSTF dated October 4, 2018, (Reference 7) provides the resolution of the concerns in terms of concerns with TSTF-286 and concerns with TSTF-51 and TSTF-471.

TSTF-286 Concerns and Resolution. Reference 7 concludes that, if a licensee includes the changes of traveler TSTF-571-T when adopting TSTF-286, the NRC staff’s technical concerns should be adequately addressed regarding TSTF-286. NSPM previously adopted TSTF-286 in PINGP TS with the conversion to improved standard TS in Unit 1 and Unit 2 Amendments 158 and 149 (Reference 5). Thus, NSPM proposes to adopt TSTF-571-T as part of the proposed amendment to address the NRC staff concern.

TSTF-51 and TSTF-471 Concerns and Resolution. Reference 7 notes that for certain facilities, LARs adopting TSTF-51 and TSTF-471 could result in exceeding the bounding licensing basis Fuel Handling Accident analysis of record dose for control room. NSPM previously adopted, in part, TSTF-51 in PINGP TS with conversion to improved standard TS (Reference 5).

Under Amendments 166 and 156 (Reference 6), NSPM adopted a portion of TSTF-51 and selectively implemented alternate source term. This included using the TSTF-51 definition of recently irradiated and changes to TS 3.3.5, Containment Ventilation Isolation Instrumentation, and TS 3.9.4, Containment Penetrations (at that time). In Reference 7, the NRC reminded licensees that an associated LAR should include information as described in the TSTF-51 Reviewer's Note. NSPM committed to the guidelines of TSTF-51 Reviewer's Note for the assessment of systems removed from service during movement of irradiated fuel at PINGP in the license amendment request for Amendments 166 and 156. Therefore, this concern has been addressed in previous approved licensing action.

Reference 7 also requested a specific discussion for any TS limiting condition for operation (LCO) which is proposed to be revised to remove the term CORE ALTERATIONS from LCO applicability. This LAR does not include removal of the term CORE ALTERATIONS from any Applicability statement, because the current PINGP TS include no Applicability statements in which the term CORE ALTERATIONS appears.

<u>TS 3.9.3, Required Action A.2</u>

The proposed change revises Required Action A.2 of TS 3.9.3 when a required core subcritical neutron flux monitor is inoperable. The proposed Required Action A.1 for TSTF-471, described above, requires immediately suspending positive reactivity additions. This action prohibits diluting the boron concentration of the coolant in the RCS, the loading of fuel assemblies or sources into the core, or the removal of reactivity control components.

The existing Required Action A.2, which prohibits introducing coolant into the RCS unless that coolant has a boron concentration greater than or equal to the boron concentration limit in LCO 3.9.1, is deleted. This action would allow dilution of the boron concentration in the RCS with one core subcritical neutron flux monitor inoperable provided the boron concentration is not reduced to less than the limit in LCO 3.9.1. Thus, removal of this action addresses the NRC staff concern that an inadvertent boron dilution could occur and the change in core reactivity may not be detected due to the inoperable core subcritical neutron flux monitor. With the removal of current Required Action A.2, the proposed Required Action A.1 would prohibit any dilution of the RCS, even if the introduced coolant has a boron concentration greater than the limit in LCO 3.9.1.

The proposed Required Action A.2 would also prohibit any movement of fuel, sources, or reactivity control components in the reactor core. With one core subcritical neutron flux monitor inoperable, the operator may not be able to monitor the core reactivity condition in part of the reactor. Therefore, the conservative action is to suspend movement of any core components that may affect reactivity until the core subcritical neutron flux monitor is restored. While unlikely, movement of fuel assemblies from one core location to another, the movement of sources, or the removal of reactivity control components, could result in an undetected change in core reactivity.

The proposed Required Action A.2 is modified by a Note that permits fuel assemblies, sources, and reactivity control components to be moved if necessary to restore an inoperable core neutron flux monitor or to complete movement of a component to a safe condition. The core subcritical neutron flux monitors are located outside the reactor core. Troubleshooting, repair, or replacement of the inoperable core subcritical neutron flux monitor may require moving fuel, sources, or reactivity control components away from the core subcritical neutron flux monitor location to minimize the radiation dose to the workers. If a fuel assembly, source, or reactivity control component is in the process of being moved when it is discovered that a required core subcritical neutron flux monitor is inoperable, the component may be placed in a safe condition.

3.3 Technical Evaluation of Administrative Change to TS 5.0 Page Numbering

This change is purely administrative and changes none of the requirements in Section 5.0.

3.4 Variations

NSPM proposes no variations from TS changes described in TSTF-571-T and is proposing the following variations from the TS changes described in the TSTF-471. These variations do not affect the applicability of TSTF-471.

- The TSTF-471 markup of NUREG-1431 standard TS 3.8.2 eliminated CORE ALTERATIONS by deleting Required Action A.2.1. The corresponding PINGP TS 3.8.2, Required Action to be deleted is Required Action A.1.
- The TSTF-471 markup of NUREG-1431 standard TS 3.8.5 eliminated CORE ALTERATIONS by deleting Required Action B.2.1. The corresponding PINGP TS 3.8.5, Required Action to be deleted is Required Action B.1.
- The TSTF-471 markup of NUREG-1431 standard TS 3.8.8 eliminated CORE ALTERATIONS by deleting Required Action A.2.1. The corresponding PINGP TS 3.8.8, Required Action to be deleted is Required Action A.1.
- PINGP TS 3.9.3, Required Action C.2 to suspend CORE ALTERATIONS when a required core subcritical neutron flux monitor audible count rate circuit is inoperable is a PINGP-specific Required Action that is not part of NUREG-1431 standard TS.
- PINGP TS do not include an LCO 3.9.2 for Unborated Water Source Isolation Valves. Therefore, the proposed change to that section of the Westinghouse Owners Group (WOG) Standard Technical Specifications (STS) proposed in TSTF-471 does not apply to PINPG and is not included in this amendment request.

The Travelers discuss the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC). PINGP was not licensed to the 10 CFR 50, Appendix A, GDC. The PINGP was designed and constructed to comply with NSP’s understanding of the intent of the AEC General Design Criteria (AEC GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Since the construction of the plant was significantly completed prior to the issuance of the February 20, 1971, 10CFR50, Appendix A GDC, the plant was not reanalyzed and the Final Safety Analysis Report (FSAR) was not revised to reflect these later criteria. However, the AEC Safety Evaluation Report acknowledged that the AEC staff assessed the plant, as described in the FSAR, against the Appendix A design criteria and “... are satisfied that the plant design generally conforms to the intent of these criteria.” This difference does not alter the conclusion that the proposed change is applicable to PINGP.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The TSs satisfy 10 CFR 50.36, "Technical specifications." The following systems and parameters meet one or more of the criteria of 10 CFR 50.36(c)(2)(ii):

- AC sources, DC sources, inverters, and distribution systems in MODES 5 or 6 or during movement of irradiated fuel assemblies.
- Boron concentration requirement in Mode 6, and
- Neutron instrumentation requirements in Mode 6.

The proposed amendment continues to provide appropriate remedial actions and shutdown requirements required by 10 CFR 50.36(c)(2)(i) for any system requiring an LCO pursuant the criteria of 10 CFR 50.36(c)(2)(ii).

The following AEC GDC are applicable to the change:

GDC 13: Fission Process Monitors and Controls. The proposed amendment does not change means provided for monitoring the fission, determining control rod position, control and determination of boron concentration.

GDC 19: Protection Systems Reliability. The proposed amendment does not change the design of protection systems.

GDC 27: Redundancy of Reactivity Control. The proposed amendment will not change the two independent reactivity control systems, rod cluster control assemblies and boric acid dissolved in the reactor coolant.

GDC 29: Reactivity Shutdown Capability. The proposed amendment will not change that the reactor may be made subcritical by the rod cluster control system sufficiently fast to prevent exceeding acceptable fuel damage limits, under all anticipated conditions even with the most reactive rod control cluster fully withdrawn.

GDC 31: Reactivity Control Systems Malfunction. The proposed amendment will not change the reactivity control systems or their ability to ensure acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

GDC 39: Emergency Power for Engineered Safety Features. The proposed amendment will not change the independent connections to the system grid or the redundant source of emergency power from the four diesel generators installed at PINGP. Power to the engineered safety features will still be assured even with the failure of a single active component in each system.

The proposed change does not affect compliance with these regulations and guidance and will insure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 Precedent

STS traveler TSTF-471 was approved for use by the NRC staff and incorporated into the applicable Standard TS NUREGs, Revision 4, published in April 2012. Some facilities have adopted, as technically practicable, TSTF-471. For example:

- Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Amendments 279 and 256.
- Point Beach Nuclear Plant, Units 1 and 2, Amendments 224 and 230.

4.3 No Significant Hazard Consideration Analysis

NSPM requests adoption of TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes," which is an approved change to the Standard Technical Specifications, into the PINGP Units 1 and 2 Technical Specifications (TS). This proposed change eliminates the defined term CORE ALTERATIONS and all uses of the term from the TS and Bases. NSPM also requests adoption of TSTF-571-T,"Revise Actions for Inoperable Source Range Neutron Flux Monitor," into the PINGP Units 1 and 2 TS. This proposed change to TS 3.9.3 ensures that no actions are taken that could alter the core reactivity when a required core subcritical neutron flux monitor is inoperable. NSPM further proposes an administrative change to reformat the page numbering of TS Section 5.0 and remove unused pages for ease of future changes.

NSPM has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No

   The proposed change to adopt TSTF-471 eliminates the use of the term CORE ALTERATIONS from TS and all Required Actions requiring suspension of CORE ALTERATIONS. CORE ALTERATIONS are not an initiator of any accident previously evaluated except a fuel handling accident. Those revised Specifications which protect the initial conditions of a fuel handling accident also require the suspension of movement of irradiated fuel assemblies. This Required Action protects the initial conditions of a fuel handling accident and, therefore, suspension of CORE ALTERATIONS is not required. Suspension of CORE ALTERATIONS does not provide mitigation of any accident previously evaluated. Therefore, CORE ALTERATIONS do not affect the initiators of the accidents previously evaluated and suspension of CORE ALTERATIONS does not affect the mitigation of the accidents previously evaluated.

   The proposed change to adopt TSTF-571-T revises the Required Action for an inoperable core subcritical neutron flux monitor to suspend the movement of fuel, sources, and reactivity control components within the reactor vessel. The Actions taken when a required core subcritical neutron flux monitor is inoperable are not initiators to any accident previously evaluated. The core subcritical neutron flux monitors are credited for detecting a boron dilution accident. The proposed change restricts the licensee's actions while a required core subcritical neutron flux monitor is inoperable beyond the current requirements, further preventing the occurrence of a boron dilution accident. Therefore, the consequences of an accident previously evaluated are not significantly increased.

The proposed administrative change to TS 5.0 page numbering and unused page removal has no impact on probability or consequences of accidents.

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

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

Response: No

No new or different accidents result from utilizing the proposed changes of TSTF-471 and TSTF-571-T. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a significant change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions.

The proposed administrative change to TS 5.0 page numbering and unused page removal has no impact on the content of TS 5.0, the plant, or plant operation; nor does it change any requirements in TS 5.0.

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

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

Response: No

Only two accidents are postulated to occur in the plant conditions in which CORE ALTERATIONS may be made: a fuel handling accident and a boron dilution accident. The proposed change to adopt TSTF-471 and suspending movement of irradiated fuel assemblies prevents a fuel handling accident. Also requiring the suspension of CORE ALTERATIONS is an overly broad, redundant requirement that does not increase the margin of safety. CORE ALTERATIONS have no effect on a boron dilution accident. Core components are not involved in the creation or mitigation of a boron dilution accident and the SHUTDOWN MARGIN limit is based on assuming the worst-case configuration of the core components. Therefore, CORE ALTERATIONS have no effect on the margin of safety related to a boron dilution accident.

The proposed adoption of TSTF-571-T revises the Required Actions for an inoperable core subcritical neutron flux monitor to prohibit the movement of fuel assemblies, sources, and reactivity control components when a required core subcritical neutron flux monitor is inoperable. No safety limits are affected. No Limiting Conditions for Operation or Surveillance limits are affected. The design, operation, surveillance methods, and acceptance criteria specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plant's licensing basis. The proposed change does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety.

The proposed administrative change to TS 5.0 page numbering and unused page removal has no impact on the content of TS 5.0 and results in no change to margin of safety.

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

Based on the above, NSPM concludes that the proposed amendment 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.

4.4 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.

5.0 ENVIRONMENTAL EVALUATION

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.

6.0 REFERENCES

1. NRC Safety Evaluation for TSTF-471-A, Revision 1, "Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes," dated December 7, 2006. (NRC ADAMS Accession No. ML062860320)

2. Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Amendments 279 and 256, dated. September 21, 2006. (NRC ADAMS Accession No. ML062350447)

3. Point Beach Nuclear Plant, Units 1 and 2, Amendments 224 and 230, dated February 15, 2007. (NRC ADAMS Accession No. ML063450073)

4. NRC letter to the Technical Specification Task Force, Potential Issues with Plant-Specific Adoption of Travelers TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,” TSTF-286, Revision 2, “Operations Involving Positive Reactivity Additions,” and TSTF-471, Revision 1, “Eliminate Use of Term Core Alterations in Actions and Notes," dated November 7, 2013. (NRC ADAMS Accession No. ML13246A358)

5. PINGP, Units 1 and 2, Amendments 158 and 149. (NRC ADAMS Accession Nos. ML022070613 and ML022070654)

6. PINGP, Units 1 and 2, Amendments 166 and 156. (NRC ADAMS Accession No. ML042430504)

7. NRC letter to Technical Specifications Task Force, Plant-Specific Adoption of Travelers TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,” TSTF-471, Revision 1, “Eliminate Use of Term CORE ALTERATIONS in Actions and Notes,” and TSTF-286, Revision 2, “Operations Involving Positive Reactivity Additions,” dated October 4, 2018. (ADAMS Accession No. ML17346A587)

ENCLOSURE, ATTACHMENT 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes"

TECHNICAL SPECIFICATION PAGES (Markup)

(56 Pages Follow)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor output as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
~~CORE ALTERATION~~	~~CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.~~
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose when inhaled as the combined activities of isotopes I-131, I-132, I-133, I-134 and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required path inoperable.	------------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. ---	
	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.~~3~~ 2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.~~4~~ 3 Initiate action to restore required path to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required DG inoperable.	~~B.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	B.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.~~3~~ 2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.~~4~~ 3 Initiate action to restore required DG to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required DC electrical power subsystem inoperable for reasons other than Condition A. OR Required Action and associated Completion Time of Condition A not met.	~~B.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	B.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.~~3~~ 2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.~~4~~ 3 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters-Shutdown

LCO 3.8.8 One Reactor Protection Instrument AC inverter shall be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.~~3~~ 2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.~~4~~ 3 Initiate action to restore required inverter to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct inverter voltage and alignment to required Reactor Protection Instrument AC panel.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems-Shutdown

LCO 3.8.10 The necessary portion of safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

--NOTE--
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required safeguards AC, DC, or Reactor Protection Instrument AC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	~~A.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~<u>AND</u>~~	
	A.2.~~2~~1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.~~3~~ 2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.~~4~~ 3 Initiate actions to restore required safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.~~5~~ 4 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling cavity shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 6.

-------------------------------------NOTE-------------------------------------
Only applicable to the refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limits.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~ 1 Suspend positive reactivity additions.	Immediately
	AND	
	A.~~3~~ 2 Initiate action to restore boron concentration to within limits.	Immediately

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two core subcritical neutron flux monitors shall be OPERABLE.

AND

One core subcritical neutron flux monitor audible count rate circuit shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required core subcritical neutron flux monitor inoperable.	A.1 Suspend positive reactivity additions ~~CORE ALTERATIONS~~. AND A.2 ------------ NOTE ------------- Fuel assemblies, sources, and reactivity control components may be moved if necessary to restore an inoperable core subcritical neutron flux monitor or to complete movement of a component to a safe condition. ---------------------------------- Suspend movement of fuel, sources, and reactivity control components within the reactor vessel. ~~Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.~~	Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required core subcritical neutron flux monitors inoperable.	B.1 Initiate action to restore one core subcritical neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours
C. Required core subcritical neutron flux monitor audible count rate circuit inoperable.	C.1 Initiate action to isolate unborated water sources.	Immediately
	~~AND~~	
	~~C.2 Suspend CORE ALTERATIONS.~~	~~Immediately~~

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active senior reactor operator (SRO) license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator (RO) license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Safety Analysis Report (USAR) or Quality Assurance Topical Report;

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2 Organization (continued)

5.2.2 Plant Staff

The plant staff organization shall include the following:

a. An operator to perform non-licensed duties shall be assigned to each reactor containing fuel and one additional operator to perform non-licensed duties shall be assigned when either or both reactors are operating in MODES 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Not Used.

5.2 Organization

5.2.2 Plant Staff (continued)

e. The operations manager or assistant operations manager shall hold an SRO license. In addition, the duty shift manager shall hold an SRO license.

f. In MODES 1, 2, 3, and 4, the shift technical advisor shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Plant Staff Qualifications

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the following:

- The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, September 1975.
- In addition, the operations manager shall be qualified as required by TS 5.2.2.e.
- The licensed operators shall comply only with the requirements of 10 CFR 55.

5.3.2 For the purpose of 10 CFR 55.4, a licensed SRO and a licensed RO are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality control for effluent and environmental monitoring;

d. Not used; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Monitoring, and Radioactive Effluent Reports required by Specification 5.6.2 and Specification 5.6.3.

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval by a member of plant management designated by the plant manager; and

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed. The date (i.e., month and year) the change was implemented shall be indicated.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The systems include portions of the Residual Heat Removal and Safety Injection Systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable.

This program shall allocate releases equally to each unit. The liquid radwaste treatment system, waste gas treatment system, containment purge release vent, and spent fuel pool vent are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. Therefore, all releases will be allocated equally in determining conformance to the design objectives of 10 CFR 50, Appendix I.

The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. for noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. for iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

5.5 Programs and Manuals

5.5.7 Inservice Testing Program (continued)

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semiquarterly	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified 2 years or less in the Inservice Testing Program for performing inservice testing activities.

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2 and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals (continued)

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5.5 Programs and Manuals (continued)

5.5.9 Ventilation Filter Testing Program (VFTP)

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

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

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

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

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

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

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

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

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

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

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

The program shall include:

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

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

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

   Condensate storage tanks
   Outside temporary tanks

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

5.5 Programs and Manuals (continued)

5.5.11 Diesel Fuel Oil Testing Program

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

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

5.5.12 Technical Specifications (TS) Bases Control Program

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

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

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

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

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

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

5.5 Programs and Manuals

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

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

5.5.13 Safety Function Determination Program (SFDP)

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

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

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

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

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

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

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

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

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

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

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

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

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

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

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

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

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

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

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

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

5.5.15 Battery Monitoring and Maintenance Program

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

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

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

5.5 Programs and Manuals (continued)

5.5.16 Control Room Envelope Habitability Program

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

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

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

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

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

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program (continued)

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

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

5.5.17 Surveillance Frequency Control Program

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

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

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

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

5.5 Programs and Manuals (continued)

5.5.18 Risk Informed Completion Time Program

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

a. The RICT may not exceed 30 days;

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

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

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

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

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

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

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

5.5 Programs and Manuals

5.5.18 Risk Informed Completion Time Program (continued)

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

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Monitoring Report

---NOTE---------------------------------------
A single submittal may be made for the plant. The submittal should combine sections common to both units.

The Annual Radiological Environmental Monitoring Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Monitoring Report shall include summarized and tabulated results, in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Report

---NOTE---
A single submittal may be made for the plant. The submittal shall combine sections common to both units.

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Not Used.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

TS 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $\left(F^N_{\Delta H}\right)$";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1;
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration".

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

1. NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology", (latest approved version);

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology";

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code";

6. Deleted;

7. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best Estimate Methodology";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II";

9. WCAP-13677-P-A, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options";

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version);

11. NAD-PI-003, "Prairie Island Nuclear Power Plant Required Shutdown Margin During Physics Tests";

12. NAD-PI-004, "Prairie Island Nuclear Power Plant $F_Q^W(Z)$ Penalty With Increasing $\left[F_Q^C(Z)/K(Z)\right]$ Trend";

13. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification";

14. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions";

15. WCAP-11397-P-A, "Revised Thermal Design Procedure";

16. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report";

17. WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod";

19. WCAP-7907-P-A, "LOFTRAN Code Description";

20. WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code";

21. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code";

22. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event";

23. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores";

24. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift";

25. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis";

26. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses";

27. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)";

28. Caldon Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

29. Caldon Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System";

30. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report";

31. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™";

32. Commencing Unit 1 Cycle 30 and Unit 2 Cycle 30, this reference shall be used in lieu of reference 23: WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON', August 2004; and

33. Commencing Unit 1 Cycle 30 and Unit 2 Cycle 30, this reference shall be used in lieu of reference 23: WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology", August 2007.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, OPPS arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) $\leq$ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

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

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6 Reporting Requirements (continued)

5.6.7 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

5.6 Reporting Requirements (continued)

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied in place of the controls required by paragraph 10 CFR 20.1601(a) and (b) of 10 CFR 20:

5.7.1 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **less than** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

5.7 High Radiation Area

5.7.1 <u>High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **less than** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates</u> (continued)

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area (continued)

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or their designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

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5.7 High Radiation Area

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source (continued)

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3) above are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device shall be used that continuously displays radiation dose rates in the area.

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5.7 High Radiation Area

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source (continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are located within a larger area where no enclosure exists for the purpose of locking and where no enclosure can be reasonably constructed around the individual area, that individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a flashing light shall be activated at the area as a warning device.

ENCLOSURE, ATTACHMENT 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes"

TECHNICAL SPECIFICATION PAGES (Re-typed)

(53 Pages Follow)

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor output as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same dose when inhaled as the combined activities of isotopes I-131, I-132, I-133, I-134 and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required path inoperable.	------------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. ---	
	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore required path to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required DG inoperable.	B.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.3 Initiate action to restore required DG to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required DC electrical power subsystem inoperable for reasons other than Condition A. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	B.3 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.5.1	---------------------------NOTE--------------------------- The following SRs are not required to be performed: SR 3.8.4.2 and SR 3.8.4.3. --- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.2 SR 3.8.4.3	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters-Shutdown

LCO 3.8.8 One Reactor Protection Instrument AC inverter shall be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

--NOTE--

LCO 3.0.3 not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.3 Initiate action to restore required inverter to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct inverter voltage and alignment to required Reactor Protection Instrument AC panel.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems-Shutdown

LCO 3.8.10 The necessary portion of safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required safeguards AC, DC, or Reactor Protection Instrument AC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.3 Initiate actions to restore required safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.4 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System and the refueling cavity shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 6.

------------------------------------NOTE------------------------------------
Only applicable to the refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limits.	A.1 Suspend positive reactivity additions. <u>AND</u> A.2 Initiate action to restore boron concentration to within limits.	Immediately Immediately

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two core subcritical neutron flux monitors shall be OPERABLE.

<u>AND</u>

One core subcritical neutron flux monitor audible count rate circuit shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required core subcritical neutron flux monitor inoperable.	A.1 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	--------------- NOTE ---------------- Fuel assemblies, sources, and reactivity control components may be moved if necessary to restore an inoperable core subcritical neutron flux monitor or to complete movement of a component to a safe condition. --	
	A.2 Suspend movement of fuel, sources, and reactivity control components within the reactor vessel.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required core subcritical neutron flux monitors inoperable.	B.1 Initiate action to restore one core subcritical neutron flux monitor to OPERABLE status. <u>AND</u> B.2 Perform SR 3.9.1.1.	Immediately Once per 12 hours
C. Required core subcritical neutron flux monitor audible count rate circuit inoperable.	C.1 Initiate action to isolate unborated water sources.	Immediately

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The shift supervisor shall be responsible for the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active senior reactor operator (SRO) license shall be designated to assume the control room command function. During any absence of the shift supervisor from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator (RO) license shall be designated to assume the control room command function.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Safety Analysis Report (USAR) or Quality Assurance Topical Report;

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2 Organization (continued)

5.2.2 Plant Staff

The plant staff organization shall include the following:

a. An operator to perform non-licensed duties shall be assigned to each reactor containing fuel and one additional operator to perform non-licensed duties shall be assigned when either or both reactors are operating in MODES 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Not Used.

5.2 Organization

5.2.2 Plant Staff (continued)

e. The operations manager or assistant operations manager shall hold an SRO license. In addition, the duty shift manager shall hold an SRO license.

f. In MODES 1, 2, 3, and 4, the shift technical advisor shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Plant Staff Qualifications

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the following:

- The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1, September 1975.
- In addition, the operations manager shall be qualified as required by TS 5.2.2.e.
- The licensed operators shall comply only with the requirements of 10 CFR 55.

5.3.2 For the purpose of 10 CFR 55.4, a licensed SRO and a licensed RO are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality control for effluent and environmental monitoring;

d. Not used; and

e. All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Monitoring, and Radioactive Effluent Reports required by Specification 5.6.2 and Specification 5.6.3.

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval by a member of plant management designated by the plant manager; and

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed. The date (i.e., month and year) the change was implemented shall be indicated.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The systems include portions of the Residual Heat Removal and Safety Injection Systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5 Programs and Manuals (continued)

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable.

This program shall allocate releases equally to each unit. The liquid radwaste treatment system, waste gas treatment system, containment purge release vent, and spent fuel pool vent are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. Therefore, all releases will be allocated equally in determining conformance to the design objectives of 10 CFR 50, Appendix I.

The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. for noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. for iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT or PT) of exposed surfaces of the removed flywheels may be conducted at 20 intervals.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

5.5 Programs and Manuals

5.5.7 Inservice Testing Program (continued)

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semiquarterly	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified 2 years or less in the Inservice Testing Program for performing inservice testing activities.

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2 and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

    a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals (continued)

5.5.9 Ventilation Filter Testing Program (VFTP)

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

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

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

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

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

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

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

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

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

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

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

The program shall include:

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

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

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

   Condensate storage tanks
   Outside temporary tanks

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

5.5 Programs and Manuals (continued)

5.5.11 Diesel Fuel Oil Testing Program

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

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

5.5.12 Technical Specifications (TS) Bases Control Program

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

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

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

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

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

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

5.5 Programs and Manuals

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

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

5.5.13 Safety Function Determination Program (SFDP)

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

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

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

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

d. Other appropriate limitations and remedial or compensatory actions.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

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

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

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

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

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

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

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

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

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

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

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

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

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

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

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

5.5.15 Battery Monitoring and Maintenance Program

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

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

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

5.5 Programs and Manuals (continued)

5.5.16 Control Room Envelope Habitability Program

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

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

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

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

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

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program (continued)

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

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

5.5.17 Surveillance Frequency Control Program

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

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

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

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

5.5 Programs and Manuals (continued)

5.5.18 Risk Informed Completion Time Program

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

a. The RICT may not exceed 30 days;

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

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

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

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

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

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

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

5.5 Programs and Manuals

5.5.18 Risk Informed Completion Time Program (continued)

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

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

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Monitoring Report

--NOTE--------------------------------------
A single submittal may be made for the plant. The submittal should combine sections common to both units.

The Annual Radiological Environmental Monitoring Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Monitoring Report shall include summarized and tabulated results, in the format of Regulatory Guide 4.8, December 1975, of all radiological environmental samples taken during the report period. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

The report shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Report

---NOTE--
A single submittal may be made for the plant. The submittal shall combine sections common to both units.

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Not Used.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   TS 2.1.1, “Reactor Core SLs”;
   LCO 3.1.1, “SHUTDOWN MARGIN (SDM)”;
   LCO 3.1.3, “Isothermal Temperature Coefficient (ITC)”;
   LCO 3.1.5, “Shutdown Bank Insertion Limits”;
   LCO 3.1.6, “Control Bank Insertion Limits”;
   LCO 3.1.8, “PHYSICS TESTS Exceptions - MODE 2”;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor $\left(F^N_{\Delta H}\right)$";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1;
LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration".

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

1. NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version);

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units" (latest approved version);

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology", (latest approved version);

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology";

5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code";

6. Deleted;

7. WCAP-10924-P-A, "Westinghouse Large Break LOCA Best Estimate Methodology";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II";

9. WCAP-13677-P-A, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO$_{TM}$ Cladding Options";

10. NSPNAD-93003-A, "Transient Power Distribution Methodology", (latest approved version);

11. NAD-PI-003, "Prairie Island Nuclear Power Plant Required Shutdown Margin During Physics Tests";

12. NAD-PI-004, "Prairie Island Nuclear Power Plant $F_Q^W(Z)$ Penalty With Increasing $\left[F_Q^C(Z)/K(Z)\right]$ Trend";

13. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification";

14. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions";

15. WCAP-11397-P-A, "Revised Thermal Design Procedure";

16. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report";

17. WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod";

19. WCAP-7907-P-A, "LOFTRAN Code Description";

20. WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code";

21. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code";

22. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event";

23. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores";

24. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift";

25. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis";

26. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses";

27. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)";

28. Caldon Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System";

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

29. Caldon Engineering Report ER-157P, “Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System”;

30. WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Reference Core Report”;

31. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™”;

32. Commencing Unit 1 Cycle 30 and Unit 2 Cycle 30, this reference shall be used in lieu of reference 23: WCAP-16045-P-A, “Qualification of the Two-Dimensional Transport Code PARAGON’, August 2004; and

33. Commencing Unit 1 Cycle 30 and Unit 2 Cycle 30, this reference shall be used in lieu of reference 23: WCAP-16045-P-A, Addendum 1-A, “Qualification of the NEXUS Nuclear Data Methodology”, August 2007.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat-up, cooldown, low temperature operation, criticality, and hydrostatic testing, OPPS arming, PORV lift settings and Safety Injection Pump Disable Temperature as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.10, "Pressurizer Safety Valves";
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) > Safety Injection (SI) Pump Disable Temperature";
LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSCLT) ≤ Safety Injection (SI) Pump Disable Temperature"; and
LCO 3.5.3, "ECCS - Shutdown".

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

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6 Reporting Requirements (continued)

5.6.7 Steam Generator Tube Inspection Report

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

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

5.6 Reporting Requirements (continued)

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied in place of the controls required by paragraph 10 CFR 20.1601(a) and (b) of 10 CFR 20:

5.7.1 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **less than** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

5.7 High Radiation Area

5.7.1 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **less than** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates (continued)

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area; or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area (continued)

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or their designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint; or

5.7 High Radiation Area

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source (continued)

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area; or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area, who is responsible for controlling personnel exposure within the area; or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3) above are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device shall be used that continuously displays radiation dose rates in the area.

5.7 High Radiation Area

5.7.2 High Radiation Areas accessible to personnel in which radiation levels could result in an individual receiving a deep dose equivalent **in excess of** 1.0 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates, but less than 500 rad in one hour at one meter from the source (continued)

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are located within a larger area where no enclosure exists for the purpose of locking and where no enclosure can be reasonably constructed around the individual area, that individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a flashing light shall be activated at the area as a warning device.

ENCLOSURE, ATTACHMENT 3

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

License Amendment Request

TSTF-471, "Eliminate Use of the Term CORE ALTERATIONS in ACTIONS and Notes"

TECHNICAL SPECIFICATION BASES PAGES (MARKUP)

FOR INFORMATION ONLY

(11 Pages Follow)

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

The Load Sequencer requirements are covered in LCO 3.3.4, "4 kV Safeguards Bus Voltage Instrumentation".

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODES 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

and

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

A required path would be considered inoperable if it were not available to at least one required Safeguards train. Although two trains may be required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of ~~CORE ALTERATIONS and~~ fuel movement.

BASES

and

ACTIONS

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

With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend ~~CORE ALTERATIONS,~~ movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

BASES

ACTIONS (continued)

A.1

Condition A represents one train with one required battery charger inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained).

Required Action A.1 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.

and

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

Condition B represents one train with one required DC electrical power subsystem inoperable for reasons other than Condition A or if the Required Actions and associated Completion Time of Condition A are not met. In this Condition there may not be adequate DC power available to support the subsystems required by LCO 3.8.10. Therefore, conservative actions are required (i.e., to suspend ~~CORE ALTERATIONS,~~ movement of irradiated fuel assemblies, and operations involving positive reactivity additions) that assure the minimum SDM or boron concentration limit is met to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

BASES

APPLICABILITY (continued)

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODES 5 and 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

and

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

If the required inverter is inoperable, the remaining OPERABLE Reactor Protection Instrument AC panel power supplies as required by LCO 3.8.10, "Distribution Systems-Shutdown," may be capable of supporting sufficient required features to allow continuation of ~~CORE ALTERATIONS,~~ fuel movement, or operations with a potential for positive reactivity additions. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend ~~CORE ALTERATIONS,~~ movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)).

BASES

APPLICABILITY (continued)	The safeguards AC, DC, and Reactor Protection Instrument AC electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9

ACTIONS

LCO 3.0.3 is not applicable while in MODES 5 and 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

and

A.1, A.2.1, A.2.2, A.2.3, A.2.4, ~~and A.2.5~~

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of ~~CORE ALTERATIONS and~~ fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend ~~CORE ALTERATIONS,~~ movement of irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to

BASES

APPLICABLE SAFETY ANALYSES (continued)	The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).
LCO	The LCO requires that a minimum boron concentration be maintained in the RCS and the refueling cavity while in MODE 6. The boron concentration limits specified in the COLR ensure that a core k_{eff} of ≤ 0.95 or other lower value is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.
APPLICABILITY	This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a k_{eff} of ≤ 0.95 or a lower value based on the dilution analysis. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical. The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling cavity when connected to the RCS. When the refueling cavity is isolated from the RCS, no potential path for boron dilution exists.
ACTIONS	<u>A.1 ~~and A.2~~</u> Continuation of ~~CORE ALTERATIONS or~~ positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS or the refueling cavity, when connected, is less than that needed to maintain shutdown margin within its limit, all operations involving ~~CORE ALTERATIONS or~~ positive reactivity additions must be

BASES

ACTIONS

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

suspended immediately.

Suspension of ~~CORE ALTERATIONS and~~ positive reactivity additions shall not preclude moving a component to a safe position.

Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.~~3~~ 2

In addition to immediately suspending ~~CORE ALTERATIONS and~~ positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

BASES (continued)

LCO

This LCO requires that two core subcritical neutron flux monitors, capable of monitoring subcritical neutron flux, be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. Neutron detectors N-31, N-32, N-51 and N-52 may be used to satisfy this LCO requirement.

This LCO also requires that one audible countrate circuit, associated with either N-31 or N-32, be OPERABLE to ensure that audible indication is available to alert the operator in containment in the event of a dilution accident ~~or improperly loaded fuel assembly~~.

APPLICABILITY

In MODE 6, the core subcritical neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, the installed detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A1 and A.2

positive reactivity additions and movement of fuel, sources, and reactivity control components within the reactor vessel

With only one required core subcritical neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, ~~CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1~~ must be suspended immediately. ~~Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.~~

BASES

ACTIONS

A.1 and A.2 (continued)

Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient (MTC), must also be evaluated to not result in reducing SDM below the required value. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

Insert A

B.1

With no required core subcritical neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a required core subcritical neutron flux monitor is restored to OPERABLE status.

B.2

With no required core subcritical neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since ~~CORE ALTERATIONS and~~ positive reactivity additions that could lead to reducing SDM below the required value are not to be made, the core reactivity condition is stabilized until the core subcritical neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

Insert A

Suspending the movement of fuel, sources, and reactivity control components ensures that positive reactivity is not inadvertently added to the reactor core while the source range neutron flux monitor is inoperable. Required Action A.2 is modified by a Note that states that fuel assemblies, sources, and reactivity control components may be moved if necessary to facilitate repair or replacement of the inoperable source range neutron flux monitor. It may be necessary to move these items away from the locations in the core close to the source range neutron flux monitor to minimize personnel radiation dose during troubleshooting or repair. The Note also permits completion of movement of a component to a safe position, should the source range neutron flux monitor be discovered inoperable during component movement.

BASES

ACTIONS (continued)

C.1 ~~and C2~~

With no audible core subcritical neutron flux monitor count rate circuit OPERABLE, only visual indication is available and prompt and definite indication of a boron dilution event would be lost. In this situation, the boron dilution event may not be detected quickly enough to assure sufficient time is available for operators to manually isolate the unborated water source and stop the dilution prior to the loss of SHUTDOWN MARGIN. Therefore, action must be taken to prevent an inadvertent boron dilution event from occurring. This is accomplished by isolating all the unborated water flow paths to the Reactor Coolant System. Isolating these flow paths ensures that an inadvertent dilution of the reactor coolant boron concentration is prevented. Since ~~CORE ALTERATIONS and~~ addition of unborated water can not be made, the core reactivity is stabilized until the audible count rate capability is restored.

The Completion Time of "Immediately" assures prompt response by operation and requires an operator to initiate actions to isolate an affected flow path immediately. Performance of Required Actions C.1 ~~and C.2~~ shall not preclude completion of movement of a component to a safe position. Once actions are initiated, they must be continued until all the necessary flow paths are isolated or the circuit is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK of required channels, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.