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10 CFR 50.90

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request (LAR) for Miscellaneous Technical Specification (TS) Improvements

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) hereby requests an amendment to the TS for the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 to make miscellaneous improvements. The proposed changes include correction of section headers in TS 1.3, "Completion Times"; removal of an SR Note in TS 3.1.4, "Rod Group Alignment Limits"; removal of applicable modes in TS 3.3.7, "Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation"; clarifications in TS 3.7.10, "Control Room Special Ventilation System (CRSVS)"; and updating a reference in TS Chapter 4.0, "Design Features". NMC has evaluated the proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

Exhibit A contains the licensee's evaluation of this LAR. Exhibit B provides a markup of TS and Bases pages. Exhibit C provides retyped TS pages.

NMC requests approval of this LAR within one calendar year of the submittal date. Upon NRC approval, NMC requests 90 days to implement the associated changes. In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR by transmitting a copy of this letter and attachments to the designated State Official.

Summary of Commitments

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

I declare under penalty of perjury that the foregoing is true and correct.
Executed on AUG 1 4 2006

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

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

Exhibits:

A. Licensee's Evaluation
B. Proposed Technical Specification and Bases Changes (markup)
C. Proposed Technical Specification Changes (retyped)

Exhibit A

LICENSEE'S EVALUATION

License Amendment Request (LAR) for Miscellaneous Technical Specification (TS) Improvements

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1.0 DESCRIPTION

This LAR is a request to amend Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2.

The Nuclear Management Company, LLC (NMC) requests Nuclear Regulatory Commission (NRC) review and approval of proposed revisions to TS 1.3, "Completion Times"; TS 3.1.4, "Rod Group Alignment Limits"; TS 3.3.7, "Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation"; TS 3.7.10, "Control Room Special Ventilation System (CRSVS)"; and TS Chapter 4.0, "Design Features". The proposed changes will improve TS usability, accuracy and conformance with the industry standard, NUREG-1431, "Standard Technical Specifications, Westinghouse Plants", Revision 3.0 (NUREG-1431).

2.0 PROPOSED CHANGE

A brief description of the associated proposed TS and Bases changes is provided below along with a discussion of the justification for each change. The specific wording changes to the TS and Bases are provided in Exhibits B and C.

> **TS 1.3, "Completion Times":** The section header at the beginning of the page text is revised to "EXAMPLE 1.3-4" since the following sentence applies to Example 1.3-4 which commences on the previous page. A new section header for Example 1.3-5 is added below the sentence. This change is acceptable because it clarifies that the sentence at the top of the page is associated with Example 1.3-4 and this change makes the PINGP TS consistent with the content guidance of NUREG-1431.

TS LCO 3.1.4, "Rod Group Alignment Limits", SR 3.1.4.1 and associated Bases: Currently the SR 3.1.4.1 Note directs the operators to Limiting Condition for Operation (LCO) 3.1.7, "Rod Position Indication", when an individual Rod Position Indication (RPI) and group demand position differ by more than 12 steps. This is confusing to the operators, since the RPI is operable when the RPI and group demand position differ by more than 12 steps if the rod position is $\leq$ 30 or $\geq$ 215 steps. This LAR proposes clarification by deleting this Note. This change does not make any technical changes which affect plant operations and is acceptable since it improves TS usability for the plant operators and conforms to the content guidance of NUREG-1431 which does not include a Note.

TS LCO 3.3.7, "Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation" and associated Bases: This license amendment proposes to remove MODES 1, 2, 3, and 4 from the Applicable Modes or Specified Conditions column in Table 3.3.7-1 for the radiation monitors which actuate the spent fuel pool special ventilation system (SFPSVS). With this change, the SFPSVS actuation instrumentation will be required to be operable during movement of irradiated fuel assemblies in the fuel pool enclosure. This change is acceptable because the proposed Applicable Modes or Specified Conditions, that is, during movement of irradiated fuel assemblies in the fuel pool enclosure, is consistent with the Applicability for the supported system in PINGP LCO 3.7.13, "Spent Fuel Pool Special Ventilation System (SFPSVS)" and the content guidance of NUREG-1431.

TS 3.7.10, "Control Room Special Ventilation System (CRSVS)": This LAR proposes to add "or B" to Condition C and revise the wording of SR 3.7.10.4.

When the Required Actions and Completion Time of Condition B are not met, Specification 3.7.10 does not provide any further guidance, thus the operators would apply LCO 3.0.3. This license amendment proposes to add "or B" to Condition C which provides shutdown requirements and is consistent with the format and content guidance of NUREG-1431. This change is acceptable since the shutdown requirements of Specification 3.7.10 Condition C are consistent with, but more conservative than the shutdown requirements of LCO 3.0.3 in that LCO 3.0.3 allows an additional hour prior to initiating actions to shutdown.

This LAR proposes to revise SR 3.7.10.4 to state, "Verify each CRSVS train in the Emergency Mode delivers 3600 to 4400 cfm through the associated CRSVS filter." This change is acceptable since it more accurately defines the required surveillance verification and thus eliminates possible confusion.

TS Chapter 4.0, "Design Features": This LAR proposes to replace Reference 1 of this TS chapter with reference to an updated document which supersedes the current reference.

The current TS 4.0 Reference 1 is a Westinghouse Electric Company (Westinghouse) Calculation Note which supports the approved spent fuel pool (SFP) criticality analyses and fuel storage configurations. This LAR proposes to replace the current reference with a reference to WCAP-16517-NP, "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", November 2005 (WCAP-16517), which superseded the Calculation Note. This change is acceptable because WCAP-16517 was used, in part, as the basis for the approval of PINGP License Amendments 172 and 162 for Units 1 and 2 respectively. Specifically, TS Figures 3.7.17-1, 4.3.1-3 and 4.3.1-4 are based on the technical content of WCAP-16517.

The Bases will also be revised where necessary to support these changes. Although the Bases changes are not a part of this LAR, marked up Bases pages are included for information.

In summary these changes are acceptable because they are conservative changes which may increase plant safety, improve TS usability or conform the TS to the format or content guidance of the applicable industry standard TS, NUREG-1431.

3.0 BACKGROUND

On July 26, 2002, the NRC issued License Amendments 158/149 which approved the Nuclear Management Company (NMC) request to convert the PINGP TS to the format and content guidance of NUREG-1431 (conversion to improved TS). With the exception of the changes proposed to TS 4.0, the TS provisions which this LAR proposes to revise were introduced in the conversion to improved TS. Through subsequent review and use of these TS, NMC has identified these improvements.

The current TS Chapter 4.0 Reference 1 was added through License Amendments 172 and 162 (LA-172/162), for Units 1 and 2 respectively, by letter, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Issuance of Amendments RE: 'Spent Fuel Pool Storage' (TAC Nos. MC5811 and MC5812)", dated February 5, 2006 (Accession No. ML060250208). During the review of the LAR which supported LA-172/162, an updated calculation was submitted to the NRC as WCAP-16517 which should be referenced in TS 4.0.

The proposed TS and Bases changes will improve the technical accuracy and usability of the TS. The proposed changes make the TS consistent with the plant design and operation or conform them to the format or content guidance of the industry standard for Westinghouse plants, NUREG-1431.

4.0 TECHNICAL ANALYSIS

PINGP is a two unit plant located on the right bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by the Northern States Power Company (NSP) and operated by NMC. Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. PINGP was not licensed to NUREG-0800, "Standard Review Plan (SRP)."

<u>Proposed TS 1.3, "Completion Times" changes</u>

TS 1.3 provides guidance on the use of Completion Times in the TS. Clarification of application of the Completion Time rules is provided through examples. The discussion of Example 1.3-4 carried over from page 1.3-10 to page 1.3-11, however, the text header on page 1.3-11 is, "EXAMPLE 1.3-5". This LAR proposes to revise the text header at the top of the page to state "EXAMPLE 1.3-4" and add new header "EXAMPLE 1.3-5" below the sentence. An extra line in the example table header is also removed. These are simple administrative changes which conforms the PINGP TS to the content and format guidance of NUREG-1431.

<u>Proposed TS LCO 3.1.4, "Rod Group Alignment Limits", SR 3.1.4.1 changes</u>

TS 3.1.4 specifies control rod group alignment requirements. TS 3.1.7, "Rod Position Indication", provides operability requirements for the control room instrumentation that indicates the location of the control rods. These two TS are closely related as discussed in TS 3.1.7 Bases:

> The operability, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available shutdown margin. Rod position indication is required to assess operability and misalignment.

The SR 3.1.4.1 Note was included as an operator aid to remind the operators to evaluate RPI readings in the context of LCO 3.1.7 as well as 3.1.4 since these two Specifications are closely related. This Note is a unique feature of the PINGP TS which

was added to the PINGP TS during conversion to ITS and is not part of the content guidance of NUREG-1431.

If the RPI differs from the group demand position by less than 12 steps, then there are no operability or alignment concerns in either LCO 3.1.4 or LCO 3.1.7. Once this difference exceeds 12 steps, then there may be operability or alignment concerns. Clearly, in LCO 3.1.7 and associated Bases, if the rod position is near the end of travel ($\leq$ 30 or $\geq$ 215 steps) then there are no operability or alignment concerns until the RPI and group demand position differ by more than 24 steps. The Background discussion in the current Bases for LCO 3.1.4 provides guidance to the operators that the actual rod position requirements of LCO 3.1.4 will be met if the RPI and group demand position are within 12 steps when the rod position is between 30 and 215 steps or they are within 24 steps when the rod position is $\leq$ 30 or $\geq$ 215 steps.

This LAR proposes to delete the SR 3.1.4.1 Note since the requirements of LCO 3.1.7 must always be met in Modes 1 and 2 when LCO 3.1.4 is applicable. In accordance with the rules of use of the TS, plant operation must meet all LCOs for which the Applicability is met. Therefore, this Note in SR 3.1.4.1 is not needed and removal of this Note will remove operator confusion caused by the Note.

This change does not affect plant operations since Specification 3.1.7 must always be met during its Modes of Applicability. This change is also consistent with the content guidance of NUREG-1431 which does not include a Note in SR 3.1.4.1. Removal of the SR 3.1.4.1 Note does not change any operating limits or requirements and the PINGP TS will continue to protect the health and safety of the public.

<u>Proposed TS LCO 3.3.7, "Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation" changes</u>

The radiation monitors required by TS 3.3.7 actuate the SFPSVS specified in TS 3.7.13, "Spent Fuel Pool Special Ventilation System (SFPSVS)". The SFPSVS ensures that radioactive materials in the fuel pool enclosure atmosphere following a fuel handling accident are filtered and adsorbed prior to exhausting to the environment. The system actuation instrumentation initiates filtered ventilation of the fuel pool enclosure automatically following receipt of a high radiation signal.

This license amendment proposes to remove Modes 1, 2, 3, and 4 from the Applicable Modes or Specified Conditions column in Table 3.3.7-1 for the radiation monitors which actuate the SFPSVS. With this change, the SFPSVS radiation monitors will be required to be operable "During movement of irradiated fuel assemblies in the fuel pool enclosure". This Applicability is independent of plant operating Modes and requires this instrumentation to be operable during any Mode for either unit whenever there is movement of irradiated fuel assemblies in the fuel pool enclosure.

This change conforms PINGP TS 3.3.7 to the content guidance of NUREG-1431. NUREG-1431 TS 3.3.8, the equivalent of PINGP TS 3.3.7, has MODES "[1,2,3,4]" in brackets to indicate that they are included on a plant specific basis if required. The

NUREG-1431 Bases B 3.3.8 discussion indicates that MODES 1, 2, 3, and 4 are included for radiation monitor actuation of FBACS (equivalent to SFPSVS at PINGP) when it is credited for filtering leakage from the Emergency Core Cooling System (ECCS) following a loss of coolant accident (LOCA). The PINGP SFPSVS is not credited for filtering post-LOCA ECCS leakage and thus, in accordance with the content guidance of NUREG-1431, MODES 1, 2, 3, and 4 should not be included in the Applicable MODES of Specified Conditions column.

The LAR which resulted in LA-166/156, issued September 10, 2004, did not credit filtration of releases from a postulated fuel handling accident. Since the PINGP TS continue to retain TS 3.7.13 and the supporting instrumentation TS 3.3.7, this LAR proposes to make the Applicability of these TS the same.

These changes make TS 3.3.7 Applicability consistent with the system design, system analyses, TS 3.7.13 which it supports, and the content guidance of NUREG-1431, thus with this change the PINGP TS will continue to protect the health and safety of the public.

Proposed TS 3.7.10, "Control Room Special Ventilation System (CRSVS)" changes

The CRSVS provides an enclosed control room environment from which the plant can be operated following an uncontrolled release of radioactivity. During normal operation, the Control Room Ventilation System provides control room ventilation. Upon receipt of an actuation signal, automatic control dampers of the associated train isolate the control room and direct a portion of recirculated air through redundant filters before entry to the air handling units.

The CRSVS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of an air handling unit, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a cleanup fan.

In addition, for the trains to be operable, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

This LAR proposes two changes to TS 3.7.10: 1) add "or B" to Condition C; and 2) revise the wording of SR 3.7.10.4.

Specification 3.7.10 Condition B allows two CRSVS trains to be inoperable due to inoperable control room boundary in Modes 1, 2, 3, or 4 for 24 hours. During that time the control room boundary must be restored to operable status. The TS do not provide further guidance if the control room boundary is not restored to operable status within 24 hours. Under the TS rules of use, when no TS guidance is provided, the operators are required to apply LCO 3.0.3 which requires the plant to initiate action within 1 hour

to shutdown, be in Mode 3 within 7 hours, be in Mode 4 within 13 hours and be in Mode 5 within 37 hours.

This proposed change provides shutdown requirements within Specification 3.7.10 which require shutdown in one hour less than the requirements of LCO 3.0.3 which the TS currently require. By adding "or B" to Condition C, the TS will provide explicit guidance for unit shutdown without applying LCO 3.0.3. The shutdown requirements of Specification 3.7.10 Condition C are more restrictive than LCO 3.0.3 in that the plant will be required to be in Mode 3 within 6 hours and Mode 5 within 36 hours. These shutdown times provide a reasonable amount of time to safely shutdown the plant. Since this proposed amendment requires the same actions within less time than LCO 3.0.3, this change will assure the plant is operated in a safe manner. This change is consistent with the content guidance of NUREG-1431 which includes "or B" in TS 3.7.10 Condition C.

Currently, SR 3.7.10.4 requires verification that "the CRSVS fan for each train delivers" the specified flow. The system design does not include any fan that is designated as "the CRSVS fan". Furthermore the system design is complex in that no single system fan by itself provides the specified flow. This LAR proposes to revise SR 3.7.10.4 to state, "Verify each CRSVS train in the Emergency Mode delivers 3600 to 4400 cfm through the associated CRSVS filter." This change does not modify any requirements for system performance; it clarifies the requirements to be technically accurate.

With these changes, operation and maintenance of the CRSVS in accordance with the PINGP TS will continue to protect the health and safety of the public.

<u>Proposed Chapter 4.0, "Design Features" changes</u>

Chapter 14, Section 4.3.1.1 provides the design basis for the SFP. Paragraphs 4.3.1.1.b and 4.3.1.1.c refer to Reference 1 as the basis for the SFP neutron multiplication factor without and with boration, respectively. The current Westinghouse calculation which is referenced supports these neutron multiplication factors, however, a more recent version of the SFP criticality analyses was developed and should be referenced.

The current Chapter 4.0 Reference 1 was added through LA-172/162. During the review of the LAR which supported LA-172/162, an updated calculation was submitted to the NRC in the form of WCAP-16517 in response to NRC requests for additional information. Although the NRC license amendment and safety evaluation did not explicitly reference WCAP-16517, the NRC did reference the supplement to the LAR dated December 2, 2005 (Accession No. ML053390121) which submitted WCAP-16517, and the NRC issued TS Figures 3.7.17-1, 4.3.1-3 and 4.3.1-4 which were derived from figures in WCAP-16517. Thus it is appropriate for Chapter 4.0, Reference 1, to be updated to reference WCAP-16517 as proposed in this LAR. This is an administrative change which does not change the basis for any current TS.

Conclusions

This LAR proposes TS changes which improve the usability and accuracy of the TS. With the exception of the SR 3.7.10.4 rewording and the TS 4.0 reference update (NUREG-1431 does not provide guidance for TS 4.0 references), the changes conform the PINGP TS to the format or content guidance of NUREG-1431. Operation and maintenance of the Prairie Island Nuclear Generating Plant with the proposed Technical Specification revisions will continue to protect the health and safety of the public.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company 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 for each of these characterizations:

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   This license amendment request proposes changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 1.3, "Completion Times", revise a text header and add a new text header; Technical Specification 3.1.4, "Rod Group Alignment Limits", remove a Surveillance Note which cross-references another Technical Specification and may cause confusion; Technical Specification 3.3.7, "Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation", revises the Modes of Applicability consistent with plant design and the Technical Specifications for the Spent Fuel Pool Special Ventilation System, the supported system; Technical Specification 3.7.10, "Control Room Special Ventilation System (CRSVS)", revises the applicability of Condition C and clarifies the requirements of the Surveillance to verify train filtration flow; and Technical Specification Chapter 4.0, "Design Features", revises Reference 1 to the most recent version of the document.

   Revising and adding text headers in Technical Specification 1.3 are administrative changes because the revised document does not change any basis for the current Technical Specifications. Since these are administrative changes, they do not involve a significant increase in the probability or consequences of a previously evaluated accident.

Technical Specification 3.1.4 assures that the control rod positions are within the limits assumed in the safety analysis and that the assumed shutdown margin is available when needed. This license amendment request proposes to remove a Note from a surveillance requirement that cross-references to Technical Specification 3.1.7. Removal of this Note does not change plant operations, testing or maintenance; therefore the proposed change does not involve a significant increase in the probability of an accident. Since plant operations, testing and maintenance are not changed, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

The Spent Fuel Pool Special Ventilation System filters radioactive materials in the fuel pool enclosure atmosphere released following a fuel handling accident. This license amendment request proposes to revise the Modes and Other Specified Conditions of Applicability for the actuation instrumentation Technical Specification to be consistent with the Modes and Other Specified Conditions of Applicability in the Technical Specification for the supported system. The Spent Fuel Pool Special Ventilation System and its actuation instrumentation are not accident initiators; therefore, the proposed changes do not affect the probability of an accident. With the proposed change, the Technical Specifications will continue to require the system actuation instrumentation to be operable when irradiated fuel is moved in the fuel pool enclosure which is also the required Applicability in the supported system Technical Specification. Since the instrumentation will be required to actuate the supported system when it is required to operate, the accident consequences will continue to be mitigated with this proposed Technical Specification change. Thus, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

The Control Room Special Ventilation System provides an enclosed control room environment from which the plant can be operated following an uncontrolled release of radioactivity. This system is not an accident initiator, thus the proposed changes do not increase the probability of an accident. This license amendment proposes changes which will: 1) reduce the time to shut down the plant when Technical Specification required actions or completion time is not met; and 2) clarifies surveillance requirements to assure that the system performs as designed. These changes do not impact the performance of the system; thus this change does not involve a significant increase in the consequences of an accident previously evaluated.

Updating the reference in Technical Specification Chapter 4.0 is an administrative change because the revised document does not change any basis for the current Technical Specifications. Since this is an administrative change, it does not involve a significant increase in the probability or consequences of a previously evaluated accident.

The changes proposed in this license amendment do not involve a significant increase the probability or consequences of an accident previously evaluated.

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

Response: No

This license amendment request proposes changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 1.3, “Completion Times”, revise a text header and add a new text header; Technical Specification 3.1.4, “Rod Group Alignment Limits”, remove a Surveillance Note which cross-references another Technical Specification and may cause confusion; Technical Specification 3.3.7, “Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation”, revises the Modes of Applicability consistent with plant design and the Technical Specifications for the Spent Fuel Pool Special Ventilation System, the supported system; Technical Specification 3.7.10, “Control Room Special Ventilation System (CRSVS)”, revises the applicability of Condition C and clarifies the requirements of the Surveillance to verify train filtration flow; and Technical Specification Chapter 4.0, “Design Features”, revises Reference 1 to the most recent version of the document.

Revising and adding text headers in Technical Specification 1.3 are administrative changes because the revised document does not change any basis for the current Technical Specifications. Since these are administrative changes, they do not create the possibility of a new or different kind of accident.

Removal of a surveillance note from Technical Specification 3.1.4 that cross-references another Technical Specification does not change any plant operations, maintenance activities or testing requirements. The Limiting Conditions for Operation will continue to be met and the proper control rod positions will continue to be maintained. There are no new failure modes or mechanisms created through the removal of the Surveillance Requirements Note, nor are new accident precursors generated by this change. This proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed revision of Modes of Applicability for the Spent Fuel Pool Special Ventilation System actuation instrumentation makes operation of the actuation instrumentation consistent with the Technical Specification requirements for the supported system and does not change the operation of the supported system for accident mitigation. The Limiting Conditions for Operation will continue to be met, no new failure modes or mechanisms are created and no new accident precursors are generated by this change. This proposed change does not create the possibility of a new or different kind of accident from any previously

evaluated.

The changes proposed for the Control Room Special Ventilation System Technical Specifications do not change any the system operations, maintenance activities or testing requirements. The Limiting Conditions for Operation will continue to be met, no new failure modes or mechanisms are created and no new accident precursors are generated by this change. This proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Updating the reference in Technical Specification Chapter 4.0 is an administrative change because the revised document does not change any basis for the current Technical Specifications. Since this is an administrative change, it does not create the possibility of a new or different kind of accident.

The Technical Specification changes proposed in this license amendment do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

This license amendment request proposes changes to the Prairie Island Nuclear Generating Plant Technical Specifications as follows: Technical Specification 1.3, "Completion Times", revise a text header and add a new text header; Technical Specification 3.1.4, "Rod Group Alignment Limits", remove a Surveillance Note which cross-references another Technical Specification and may cause confusion; Technical Specification 3.3.7, "Spent Fuel Pool Special Ventilation System (SFPSVS) Actuation Instrumentation", revises the Modes of Applicability consistent with plant design and the Technical Specifications for the Spent Fuel Pool Special Ventilation System, the supported system; Technical Specification 3.7.10, "Control Room Special Ventilation System (CRSVS)", revises the applicability of Condition C and clarifies the requirements of the Surveillance to verify train filtration flow; and Technical Specification Chapter 4.0, "Design Features", revises Reference 1 to the most recent version of the document.

Revising and adding text headers in Technical Specification 1.3 are administrative changes because the revised document does not change any basis for the current Technical Specifications. Since these are administrative changes, they do not involve a significant reduction in a margin of safety.

Plant operations are required to meet all Technical Specifications for which the Applicability is met; therefore, removal of the cross-reference Note from a

Technical Specification 3.1.4 surveillance requirement does not change how the plant is operated and therefore, this change does not involve a significant reduction in a margin of safety.

Technical Specification 3.3.7 provides requirements for actuation instrument which supports the operation of the Spent Fuel Pool Special Ventilation System as required by Technical Specification 3.7.13. The current Applicability for Technical Specification 3.3.7 requires the actuation instrumentation to be operable in Modes which are not required by Technical Specification 3.7.13. This license amendment proposes to make Technical Specification 3.3.7 Applicability the same as Technical Specification 3.7.13. This change does not reduce the conditions or Modes when the Spent Fuel Pool Special Ventilation System will operate and perform its accident mitigation function; thus this change does not involve a significant reduction in a margin of safety.

This license amendment proposes changes to the Control Room Special Ventilation System Technical Specifications which will: 1) reduce the time to shut down the plant when Technical Specification required actions or completion time is not met; and 2) clarifies surveillance requirements to assure that the system performs as designed. The proposed time to shut down the plant is consistent with other Technical Specifications for shutting down the plant and allows adequate time for an orderly shut down of the plant; thus this change does not involve a significant reduction in a margin of safety. The surveillance requirement clarifications do not reduce any testing requirements and will continue to demonstrate that the system can perform its required safety function and satisfy the Limiting Conditions for Operation. Thus this change does not involve a significant reduction in a margin of safety.

Updating the reference in Technical Specification Chapter 4.0 is an administrative change because the revised document does not change any basis for the current Technical Specifications. Since this is an administrative change, it does not involve a significant reduction in a margin of safety.

The Technical Specification changes proposed in this license amendment do not involve a significant reduction in a margin of safety.

Based on the above, the Nuclear Management Company 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.

5.2 Applicable Regulatory Requirements/Criteria

Title 10 Code of Federal Regulations 50.36, "Technical specifications":

(c) Technical specifications will include items in the following categories:

(3) *Surveillance requirements*. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

This license amendment request proposes to remove a Note from a Technical Specification 3.1.4 surveillance requirement that cross-references to Technical Specification 3.1.7. With this change, the surveillance requirement will continue to assure that the control rod alignment requirements are met and thus the facility will be within the safety limits and the limiting conditions for operation will be met.

This license amendment request proposes to clarify Control Room Special Ventilation System surveillance requirement terminology to assure that the system performs as designed. With this change the surveillance requirement will continue to assure the system is maintained and that the limiting conditions for operation will be met.

Thus with the changes proposed in this license amendment request, the requirements of Title 10 Code of Federal Regulations 50.36 continue to be met and the plant Technical Specifications will continue to provide the basis for safe plant operation.

General Design Criteria

The construction of the PINGP was significantly complete prior to issuance of 10 CFR 50, Appendix A, General Design Criteria. The PINGP was designed and constructed to comply with the Atomic Energy Commission General Design Criteria as proposed on July 10, 1967 (AEC GDC) as described in the plant Updated Safety Analysis Report (USAR). AEC GDC proposed criteria 6, 14, 27 and 28 provide guidance applicable reactor reactivity and power distribution design requirements. AEC GDC proposed criterion 66 provides guidance for spent fuel pool criticality prevention. Additionally, PINGP is committed to maintain the dose to the control room operators less than the limits specified in 10 CFR 50, Appendix A, GDC 19.

AEC GDC Criterion 6 - Reactor Core Design

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to

recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

AEC GDC Criterion 14 - Core Protection Systems

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

AEC GDC Criterion 27 - Redundancy Of Reactivity Control

At least two independent reactivity control systems, preferably of different principles, shall be provided.

AEC GDC Criterion 28 - Reactivity Hot Shutdown Capability

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

This license amendment request proposes to remove a Note from a surveillance requirement in the Technical Specification for Rod Group Alignment Limits that cross-references to the Technical Specification for Rod Position Indication. With this change, the AEC GDC stated above will continue to be met when the plant is operated with the plant Technical Specifications revised as proposed. Thus with the changes proposed in this license amendment request, the requirements of AEC GDC 6, 14, 27 and 28 continue to be met and the plant Technical Specifications will continue to provide the basis for safe plant operation.

AEC GDC Criterion 66 – Prevention of Fuel Storage Criticality

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configuration shall be emphasized over procedural controls.

Physical systems require analyses, including criticality analyses, to demonstrate that they provide a safe configuration. The specific neutron multiplication factor, k_{eff}, criteria which the criticality analyses are required to meet to demonstrate compliance with AEC GDC 66 for the spent fuel pools are included in TS Section 4.3.1.1 paragraphs b. and c. These paragraphs reference the criticality analyses to provide the basis for the neutron multiplication factor. This license amendment proposes to update the reference to the most recent version contained in WCAP-16517 which was previously submitted to the NRC for review on December 2, 2005. This license amendment does not propose to change the neutron multiplication factors, nor does it change any methodology or physical plant configurations. Thus with the change proposed in this license

amendment, the requirements of AEC GDC 66 continue to be met and the plant Technical Specifications will continue to provide the basis for safe plant operation.

10 CFR 50 Appendix A, Criterion 19 - Control room.

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

This license amendment request proposes to add shutdown requirements to the Technical Specifications for the Control Room Special Ventilation Special Ventilation System when the control room ventilation boundary is inoperable, and clarify surveillance requirements for the Control Room Special Ventilation Special Ventilation System. The proposed Technical Specification changes will assure that the Control Room Special Ventilation System continues to maintain the dose to the control room operators less than the limits specified in 10 CFR 50, Appendix A, GDC 19 under accident conditions. Thus with the changes proposed in this license amendment request, the limits of 10 CFR 50 Appendix A GDC 19 will be met and the plant Technical Specifications will continue to provide the basis for safe plant operation.

NUREG-1431 Standard Technical Specifications, Westinghouse Plants, Revision 3.0

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3.0 (NUREG-1431) provides guidance for Technical Specifications for plants with Westinghouse Nuclear Steam Supply Systems and has been approved for use by the Nuclear Regulatory Commission. The proposed Technical Specification changes are consistent with the guidance of NUREG-1431 as follows.

NUREG-1431 Example 1.3-4 concludes with the sentence, "If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered." Thus the proposal to revise the existing text header to read "EXAMPLE 1.3-4" above this sentence and a new "EXAMPLE 1.3-5" text header below this sentence is consistent with NUREG-1431 content guidance.

NUREG-1431 Specification 3.1.4, Surveillance Requirement 3.1.4.1 does not include any Notes, thus the proposal to delete the Note from Prairie Island Specification 3.1.4 is consistent with NUREG-1431 guidance.

NUREG-1431 Specification 3.3.8, the equivalent to Prairie Island Specification 3.3.7, includes brackets around Modes "1, 2, 3, and 4" in the Applicability for the Fuel Building Radiation monitors which indicates specification of these Modes in the Applicability is optional dependent on the plant licensing basis. Further guidance is provided in NUREG-1431 Bases 3.3.8 which also shows these Modes bracketed and states, "The automatic FBACS actuation instrumentation is also required in MODES [1, 2, 3, and 4] to remove fission products caused by post LOCA [loss of coolant accident] Emergency Core Cooling Systems leakage." The Prairie Island Nuclear Generating Plant Spent Fuel Pool Special Ventilation System is not credited in the safety analyses for operation following a loss of coolant accident, thus operability during Modes 1, 2, 3, and 4 is not required. Removal of Modes 1, 2, 3, and 4 from the Applicability of Prairie Island Nuclear Generating Plant Specification 3.3.7 for the Fuel Pool Enclosure Radiation is consistent with the guidance of NUREG-1431 and the Prairie Island Nuclear Generating Plant licensing basis.

NUREG-1431 Specification 3.7.10, Condition C references Condition B as a Condition for which Condition C is applicable. Thus the proposal to reference Condition B in Prairie Island Specification 3.7.10 Condition C as a Condition for which it is applicable is consistent with NUREG-1431 guidance.

NUREG-1431 does not provide guidance for the use of references in Chapter 4.0, "Design Features".

Thus, with the changes proposed in this license amendment request, the format and content guidance NUREG-1431 is met as discussed above and the plant Technical Specifications will continue to provide the basis for safe plant operation.

<u>Regulatory Requirements/Criteria Conclusions</u>

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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment 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 amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed

amendment 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 amendment.

Exhibit B

Proposed Technical Specification and Bases Changes (markup)

Technical Specification Pages

1.3-11
3.1.4-4
3.3.7-4
3.7.10-2
3.7.10.3
4.0-4

Bases pages
(for information only)

B 3.1.4-11
B 3.3.7-2

8 pages follow

1.3 Completion Times

EXAMPLES ~~(continued)~~

EXAMPLE 1.3-4~~5~~ (continued)

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

EXAMPLE 1.3-5

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	~~NOTE~~ ~~If RPI differs by > 12 steps from the group step counter demand position, enter LCO 3.1.7 to determine RPI OPERABILITY.~~ Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod, not fully inserted in the core, $\geq$ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. Both reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

Table 3.3.7-1 (page 1 of 1)
SFPSVS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Fuel Pool Enclosure Radiation	~~1,2,3,4~~ (a)	1 per train	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.3	(b)

(a) During movement of irradiated fuel assemblies in the fuel pool enclosure.
(b) This value provided by the ODCM.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CRSVS train in operation. OR D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. Two CRSVS trains inoperable during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CRSVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRSVS train ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CRSVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRSVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Verify each~~the~~ CRSVS train in the Emergency Mode~~fan in each train~~ delivers 3600 to 4400 cfm through the associated CRSVS filter.	24 months on a STAGGERED TEST BASIS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies not including those assemblies which can be returned to the reactor. The southeast corner of the small pool serves as the spent fuel cask lay down area. To facilitate plant evolutions, four additional storage racks, with a combined capacity of 196, may be temporarily installed in the cask lay down area to provide a total of 1582 storage locations (Ref. 3).

REFERENCES

1. "Prairie Island Units 1 and~~&~~ 2 Spent Fuel Pool Criticality Analysis", WCAP-16517-NP, Revision 0~~Calculation Note Number CN-WFE-03-40~~, Westinghouse Electric Company, November 2005~~11, 2004~~.

2. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

3. USAR, Section 10.2.

BASES

ACTIONS (continued)

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. The unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

~~SR 3.1.4.1 is modified by a Note which direct the operators to Specification 3.1.7, "Rod Position Indication," if a rod appears to be misaligned by more than 12 steps. If the rod position indication is determined to be correct in accordance with Specification 3.1.7, then the operator must return to Specification 3.1.4 and enter the appropriate Conditions for rod misalignment.~~

BASES (continued)

LCO

The LCO requirements ensure that instrumentation necessary to initiate the SFPSVS is OPERABLE.

1. <u>Fuel Pool Enclosure Radiation</u>

The LCO specifies two required Radiation Monitor channels (R-25 and R-31) to ensure that the radiation monitoring instrumentation necessary to initiate the SFPSVS remains OPERABLE.

The allowable value for these radiation monitors is provided by the Prairie Island Offsite Dose Calculation Manual (ODCM).

APPLICABILITY

High radiation initiation of the SFPSVS must be OPERABLE ~~in MODES 1, 2, 3, and 4 and~~ when moving irradiated fuel assemblies in the fuel pool enclosure, to ensure automatic initiation of the SFPSVS to remove fission products associated with ~~leakage after~~ a fuel handling accident.

~~While in MODES 5 and 6 without fuel handling in progress, the SFPSVS instrumentation need not be OPERABLE since a fuel handling accident cannot occur.~~

Exhibit C

Proposed Technical Specification and Bases Changes (retyped)

Technical Specification Pages

1.3-11
3.1.4-4
3.3.7-4
3.7.10-2
3.7.10-3
4.0-4

6 pages follow

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-4 (continued)

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

EXAMPLE 1.3-5

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod, not fully inserted in the core, $\geq$ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 1.8 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. Both reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

Table 3.3.7-1 (page 1 of 1)
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(a) During movement of irradiated fuel assemblies in the fuel pool encbsure.
(b) This value provided by the ODCM.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CRSVS train in operation. OR D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. Two CRSVS trains inoperable during movement of irradiated fuel assemblies.	E.1 Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CRSVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRSVS train ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CRSVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRSVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.10.4	Verify each CRSVS train in the Emergency Mode delivers 3600 to 4400 cfm through the associated CRSVS filter.	24 months on a STAGGERED TEST BASIS

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies not including those assemblies which can be returned to the reactor. The southeast corner of the small pool serves as the spent fuel cask lay down area. To facilitate plant evolutions, four additional storage racks, with a combined capacity of 196, may be temporarily installed in the cask lay down area to provide a total of 1582 storage locations (Ref. 3).

REFERENCES

1. "Prairie Island Units 1 and 2 Spent Fuel Pool Criticality Analysis", WCAP-16517-NP, Revision 0, Westinghouse Electric Company, November 2005.

2. "Criticality Analysis of the Prairie Island Units 1 & 2 Fresh and Spent Fuel Racks", Westinghouse Commercial Nuclear Fuel Division, February 1993.

3. USAR, Section 10.2.