JAN 2 9 2007

L-PI-06-097
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) to Revise Technical Specification (TS) 3.5.3 Operability Requirements for Safety Injection (SI) Subsystem

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 revise TS 3.5.3, "ECCS (Emergency Core Cooling Systems) – Shutdown" operability requirements for the SI subsystem. 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 TS 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 exhibits 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 JAN 2 9 2007

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

(copies and exhibits listed on page 2)

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) to Revise Technical Specification (TS) 3.5.3 Operability Requirements for Safety Injection (SI) Subsystem

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 3.5.3, "ECCS (Emergency Core Cooling Systems) – Shutdown" operability requirements for the SI subsystem. The proposed changes will allow the required SI pump to be rendered incapable of injecting into the Reactor Coolant System (RCS) during low temperature (MODE 4) operations due to a single action or automatic signal. The capability of the plant operators to initiate SI flow on a timely basis will be maintained and plant safety may be improved with the SI pump incapable of inadvertently injecting into the RCS during low temperature operations.

2.0 PROPOSED CHANGE

A brief description of the associated proposed TS and TS 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 LCO 3.5.3, "ECCS - Shutdown", LCO Notes and associated Bases:** This LAR proposes to modify the Limiting Condition for Operation (LCO) statement by adding a Note which allows an SI train to be considered OPERABLE when the SI pump is incapable of injecting into the RCS through a single action or automatic signal. SI flow can be initiated through manual operator actions. The proposed change to LCO 3.5.3 will clarify TS requirements with respect to NRC guidance previously provided to PINGP for the SI pumps in this mode. This change is

acceptable because there are no TS required instruments that initiate automatic actuation during the Applicability of this TS, that is, operator actions are currently assumed by this TS for the safety function to be performed. The proposed changes are also consistent with the existing LCO Note that allows a residual heat removal (RHR) train to be considered OPERABLE when it is aligned for decay heat removal and requires operator actions for the subsystem to perform its function. The Bases will also be revised as 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 consistent with the current licensing basis for this TS.

3.0 BACKGROUND

The PINGP currently operates with the reactor coolant system (RCS) water solid when below 350 °F to minimize the thermal gradient between the RCS and the pressurizer. An inadvertent mass input into the RCS from an SI pump would require the low temperature overpressure (LTOP) system to mitigate an RCS overpressure event and may also result in the loss of decay heat removal for some time until restored by operator action.

The current Bases state that manual actions to initiate emergency core cooling are acceptable in MODE 4 due to the reduced RCS pressures and temperatures. In a letter dated October 16, 1997 (Reference 1), the NRC stated that it was acceptable to have both SI pumps in "pullout" for a portion of MODE 4, that is, the switch handle is pulled out of the switch and locked in this position which prevents automatic starting of the pump. This LAR proposes TS changes which will clarify the TS requirements for SI pump status in MODE 4 and improve plant safety by further reducing the likelihood of a low temperature overpressure event and the concomitant loss of residual heat removal system (RHR) decay heat removal. With these TS changes the plant will continue to operate safely and the health and welfare of the public is protected.

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)."

Emergency Core Cooling System (ECCS) Technical Description

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Loss of secondary coolant accident, including uncontrolled steam release; and

c. Steam generator tube rupture (SGTR).

The ECCS consists of two separate subsystems: SI and RHR, with each subsystem consisting of two redundant, 100% capacity trains.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the RCS following the accidents listed above. The major components of each subsystem are the RHR pumps, RHR heat exchangers, and the SI pumps. Both subsystems consist of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core if necessary due to individual component inoperability.

The primary purpose of the SI system is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and ensures that the core will remain substantially intact and in place, with its heat transfer geometry preserved.

Current TS Requirements, Basis and Limitations

Current TS 3.5.3, "ECCS-Shutdown", requires one ECCS train to be operable when operating within the Applicability which is "MODE 4 when both RCS cold leg temperatures are greater than the SI pump disable temperature specified in the PTLR (Pressure and Temperature Limits Report)". Currently this Specification applies for RCS temperatures between 350 °F, the defined upper temperature limit for MODE 4, and 218 °F, the current SI pump disable temperature specified in the PTLR.

During low temperature conditions in the RCS, limitations are placed on the maximum number of SI pumps that may be capable of injecting into the RCS to assure the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. RCS overpressure protection is provided by: 1) restricting coolant input capability; and 2) ensuring adequate pressure relief capacity.

1) To limit coolant input capability, TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) > Safety Injection (SI) Pump Disable Temperature," requires that at least one SI pump is incapable of injecting into the RCS during the LCO Applicability. The Applicability is specified as MODE 4 when any RCS cold leg temperature is less than or equal to the overpressure protection system (OPPS) enable temperature specified in the PTLR (currently specified as 310 °F) and above the SI pump disable temperature (currently specified in the PTLR as 218 °F).

2) To provide adequate pressure relief capacity in MODE 4 within the Applicability temperature limits (between 310 °F and 218 °F), one pressurizer power operated relief valve (PORV) is the overpressure protection device that acts to terminate an increasing pressure event. Two PORVs are required for redundancy.

The pressurizer PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room. Each PORV is also equipped with an air accumulator to open the PORV in the event the normal air supply is unavailable. During low temperature operations, the OPPS provides the open signal to the PORVs. The OPPS monitors both RCS temperature and RCS pressure and indicates when a condition not acceptable in the PTLR is approached.

During plant operations in Mode 4, when the RCS temperature is below 350 °F, the RCS transitions to a water solid condition, that is, there is no steam space in the pressurizer. The plant is operated in this manner to minimize the thermal gradient between the RCS and the pressurizer. In the water solid condition, a sudden mass input, such as an inadvertent SI pump start will cause an RCS overpressure event that would rely upon the LTOP system to mitigate and may also result in loss of RHR decay heat removal for some time until restored by operator action. Decay heat removal will be affected because the RCS to RHR valves may close. Although the probability of an inadvertent SI pump start is low, the probability of this event can easily be further reduced with the TS changes proposed in this LAR. As discussed below, the plant licensing basis will continue to be met with the proposed TS changes.

This LAR proposes to make both SI pumps incapable of injecting into the RCS through a single action or automatic signal and provide additional plant safety through reduction of the probability of inadvertent SI pump start. PINGP was previously given guidance by the NRC that the SI pumps are operable for at least part of Mode 4 when their

switches are in pullout. In Reference 1, the NRC stated:

> The staff agrees with NSP[1] that the SI system is OPERABLE and there is no violation of the TS between reactor coolant system (RCS) temperatures of 310°F and 200°F[2] with both SI control switches in the pullout position, and that the operator has a reasonable amount of time to perform the actions necessary to place the control switch in the pullout or AUTO position at the RCS transition temperatures of 310 °F and 200 °F. However, the staff requests that NSP submit a TS change to eliminate any ambiguity or question regarding OPERABILITY during changing plant conditions.

This LAR proposes to eliminate any ambiguity or questions with respect to SI system operability for the whole Applicability of TS 3.5.3, that is, from 350 °F to 218 °F with both SI pumps incapable of injecting into the RCS through a single action or automatic signal.

Proposed TS 3.5.3, "ECCS – Shutdown," LCO Note changes

This LAR proposes to add a Note which modifies the LCO 3.5.3, "ECCS – Shutdown" by stating, "An SI train may be considered OPERABLE when capable of being manually aligned for ECCS injection." Clarification would also be added to the Bases LCO discussion as shown in Exhibit B.

The proposed TS and Bases provisions will allow the plant operators to make both trains of SI incapable of injecting into the RCS through a single action or automatic signal. This may be accomplished by placing the SI pump switches in pullout with a blocking device installed over the control switch that would prevent an unplanned pump start as currently specified in the Bases for Surveillance Requirement (SR) 3.4.12.1 and 2, and SR 3.4.13.1 and 2. As a requirement of the proposed TS and Bases, the plant operators must maintain the capability to manually align the SI pumps on a timely basis for injection into the RCS.

Basis for Proposed TS Revisions

The ECCS provides core cooling and negative reactivity to ensure the core is protected following a design basis accident. According to the Bases for TS 3.5.3, due to the lower RCS temperatures and pressures in Mode 4, the probability of occurrence of a design basis accident is reduced and therefore, the ECCS operational requirements are reduced. The RHR system may be aligned to provide normal shutdown cooling during Mode 4 and thus time may be required for manual alignment to the ECCS mode. The

[1] Northern States Power Company (NSP) is the owner of PINGP and was the operating entity until August 7, 2000.
[2] The TS lower temperature limit for one SI pump switch not in pullout was 200 °F in October 1997 in accordance with License Amendments 127 (Unit 1) and 119 (Unit 2). The current TS lower temperature limit is the "SI pump disable temperature specified in the PTLR" which is currently 218 °F.

Bases for PINGP TS 3.5.3 state:

> In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA [design basis accident]. Therefore, only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered for this LCO due to the time available for operators to respond to an accident.

Similar wording is also contained in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3.1 (NUREG-1431) provides format and content guidance for Technical Specifications for plants with Westinghouse Nuclear Steam Supply Systems and has been approved for use by the Nuclear Regulatory Commission.

Due to the lower RCS temperatures and pressures in Mode 4, events would evolve more slowly such that sufficient time exists for manual alignment of the RHR subsystem. Likewise, sufficient time exists for manual alignment of the SI subsystem.

Currently, the TS 3.5.3 LCO statement is modified by an LCO Note which explicitly states that the RHR subsystem may be aligned for decay heat removal, which means it is not capable of injecting as an ECCS, providing it is capable of being manually realigned to the ECCS mode of operation. The slower evolution of plant events in MODE 4 also allows time for manual alignment of the SI subsystem to inject into the RCS. This LAR proposes to add a Note which modifies the LCO statement to consider SI pumps operable when capable of being manually aligned for ECCS injection. This follows the example of the RHR subsystem which, in accordance with the current LCO Note, may be considered operable when capable of manual alignment to the ECCS mode of operation.

Although the current TS could be interpreted to require SI alignment for automatic injection, there is no instrumentation required to be operable to initiate automatic injection. TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation", Table 3.3.2-1 provides the requirements for ECCS actuation instrumentation, specifically for the SI pump (at PINGP there are no automatic actuation functions for the RHR subsystem). In Table 3.3.2-1 there are only two SI functions required to be operable in Mode 4 which are Function 1.a, "Manual Initiation", and Function 1.b, "Automatic Actuation Relay Logic". These two functions do not automatically initiate SI but allow manual initiation; thus, there are no system process instruments required to be operable that would detect system conditions for initiation of SI. The basis for the operability requirement for these two functions is discussed in Bases 3.3.2 as follows:

> Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a SI, actuation is simplified by the use of the manual

> actuation switches. Automatic actuation relay logic must be OPERABLE in MODE 4 to support system level manual initiation.

Thus, maintaining the PI system in a status capable of automatic initiation does not serve a useful function since there are no viable instruments to initiate automatic injection. As a corollary to this design feature, the plant design basis assumes operator manual actions to mitigate events which require the use of SI. With the changes proposed in this LAR, the Manual Initiation and Automatic Actuation Relay Logic Functions will continue to be operable and capable of performing their required functions when the operator returns the SI pump switch to the "Auto" position.

By letter dated September 12, 1997, NSP requested a TS interpretation for SI system operability during movement into different plant conditions. In response to this request, the NRC stated in the cover letter for Reference 1, "The [NRC] staff agrees with NSP that the SI system is OPERABLE and there is no violation of the TS between reactor coolant system (RCS) temperatures of 310 °F and 200 °F with both SI control switches in the pullout position . . ." The discussion in the attachment to Reference 1 stated,

> For changing plant conditions at 200 °F, NSP has re-interpreted the TS to allow both SI pump control switches to be in pullout position. This is based on the interpretation that both TS 3.3.A.3 and 3.3.A.4 requirements can be met if the TS phrase "at least one" is interpreted to allow both SI pump control switches to be placed in pullout prior to cooling down to 200 °F. Thus, one switch will be placed in pullout when the RCS is below 310 °F until the temperature approaches 200 °F. At that time both switches will be placed in pullout. This maintains the plant in a safe condition since events that require operation of the SI system at low RCS temperatures occur slowly and allow ample time for the operator to manually place the SI system in service. The staff agrees that the phrase "at least one" will allow both SI control switches to be placed in the pullout position at any time the RCS temperature is < 310 °F.

This LAR seeks to extend these conclusions to the full range of TS 3.5.3 Applicability, that is, Mode 4 above the SI disable temperature (from 350 °F to 218 °F). For the purposes of this consideration, the conditions from 350 °F to 310 °F do not significantly differ from the conditions between 310 °F and 200 °F. The performance of the plant over these temperature ranges is similar in that the reactor is subcritical (decay heat and the reactor coolant pumps are the primary sources of RCS heat), temperatures and pressures are relatively low, events would evolve slowly, operators have time to take manual actions to mitigate events and there is no required operable SI automatic actuation instrumentation.

An SI train can be made incapable of automatic injection by placing the switch in pullout. To prevent operation due to a single operator action the switch is covered with a blocking device. With the switch in the pullout position with a blocking device installed over the control switch, the SI pump is not capable of starting due to any single action or automatic signal. Placement of a blocking device over the switch provides additional

assurance that the SI pump switch position will not be inadvertently changed by an operator. If the SI pump is required to support plant operations, the device is simply lifted off the switch, the handle returned to the operating position and the switch is turned to the pump run position. This method of assuring that an SI pump is not inadvertently started is currently specified in the Bases for SR 3.4.12.1 and 2, and SR 3.4.13.1 and 2.

The ECCS is designed to mitigate a DBA which initiates during Mode 1, 2 or 3 depending on which provides the bounding plant operating conditions. Due to rapid development of the DBA in these Modes, the ECCS is automatically initiated to assure the assumed mitigation occurs. In Mode 4 the RCS temperatures and pressures are less than those assumed in the DBA and thus, there is time for manual initiation of the ECCS and there are no TS required functions which automatically initiate ECCS. Because the plant design, TS and Bases assume operator action during Mode 4 initiation of SI, the changes proposed in this LAR do not introduce new operator actions. Removal of the blocking device and returning the switch handle to the normal position are actions that take a few seconds and do not hinder intentional operator actions. The purpose of these measures is to prevent inadvertent operator actuation of the SI pumps by requiring these additional actions.

The NRC, in Reference 1, requested NSP to submit a TS change to eliminate ambiguity or questions regarding SI operability during changing plant conditions. This LAR proposes TS changes which implement the NRC request. Without the changes proposed in this LAR, during a plant cool down the control room operators are currently required to focus specifically on the RCS temperature as it approaches the SI pump disable temperature specified in the PTLR (currently 218 °F) so that the pump switch can be placed in pullout just prior to decreasing below the SI pump disable temperature. With the TS changes proposed in this LAR, plant safety may be improved by placing the SI pump switches in pullout when MODE 4 is entered and freeing operators to focus on other plant conditions as the plant cools down.

Conclusions

This LAR proposes TS changes which will prevent automatic or inadvertent operation of an SI pump in Mode 4 by allowing its switch to be placed in the pullout position. These changes will maintain SI operability and availability for manual actuation if required to mitigate an accident. 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 to add a new Note to Technical Specification 3.5.3, "Emergency Core Cooling System – Shutdown". This Note will allow the Safety Injection system to be considered operable within the Limiting Condition for Operation requirements while the system is not capable of automatic injection provided it is capable of being manually aligned for injection.

   This Emergency Core Cooling System is not an accident initiator, thus the proposed changes do not increase the probability of an accident. The current licensing basis, Technical Specifications and Bases do not require automatic initiation instrumentation for the Emergency Core Cooling System in Mode 4, but rather assume operator action to mitigate an accident. With the proposed Technical Specification and Bases changes, the Emergency Core Cooling System will continue to be operable for manual initiation. Since the changes proposed in this license amendment request do not impact the performance of the system, these changes do not involve a significant increase in the consequences of an accident previously evaluated.

   The changes proposed in this license amendment do not involve a significant increase in 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 to add a new Note to Technical Specification 3.5.3, "Emergency Core Cooling System – Shutdown". This Note will allow the Safety Injection system to be considered operable within the Limiting Condition for Operation requirements while the system is not capable of automatic injection provided it is capable of being manually aligned for injection.

   The changes proposed for the Emergency Core Cooling System Technical

Specifications do not change any system operations, maintenance activities or testing requirements. The Limiting Condition 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. 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 to add a new Note to Technical Specification 3.5.3, "Emergency Core Cooling System – Shutdown". This Note will allow the Safety Injection system to be considered operable within the Limiting Condition for Operation requirements while the system is not capable of automatic injection provided it is capable of being manually aligned for injection.

The current licensing basis, Technical Specifications and Bases rely upon operator actions to initiate safety injection to mitigate an accident in Mode 4 and do not require operability of any process instrumentation capable of automatically initiating the Emergency Core Cooling System. With the changes proposed in this license amendment request, the safety injection system will continue to be operable and the plant will continue to rely on operator actions for safety injection initiation. Thus, the Technical Specification changes proposed in this license amendment request 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:

(2) *Limiting conditions for operation.* (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(C) *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

This license amendment request proposes to add a Limiting Condition for Operation Note in Technical Specification 3.5.3, "ECCS [Emergency Core Cooling Systems] – Shutdown" which will allow the safety injection system to be considered operable when it is incapable of automatic operation in Mode 4 if capable of being manually aligned for ECCS injection. The ECCS in Mode 4 meets the requirements of Title 10 Code of Federal Regulations 50.36(c)(2)(ii)(C) and thus is included in the Prairie Island Nuclear Generating Plant Technical Specifications. With the addition of the proposed Limiting Condition for Operation Note, Technical Specification 3.5.3 continues to provide "the lowest functional capability . . . of equipment required for safe operation of the facility". The safety injection subsystem of ECCS will continue to be operable and capable of being manually aligned for mitigation of an accident. The safety injection system will not be required to be capable of automatic injection for accident mitigation. However, in Mode 4, the plant licensing basis, the Technical specifications and Bases rely upon operator actions to initiate safety injection due to the lower temperatures and pressures in the reactor coolant system. For the same reason, the current Technical Specifications do not require automatic safety injection actuation instruments to be operable in Mode 4. With the changes proposed in this license amendment request, safety injection system operability is maintained and the limiting condition 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 criterion 37 provides design guidance for the operating capability of the Safety Injection system.

AEC GDC Criterion 37 - Engineered Safety Features Basis For Design

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

This license amendment request proposes to add a Limiting Condition for Operation Note in Technical Specification 3.5.3, "ECCS – Shutdown" that allows a safety injection pump to be considered operable for the purposes of this Specification when the pump is incapable of automatic initiation. With this proposed change, the safety injection subsystem of the Emergency Core Cooling System will continue to be a back up safety system that is capable of mitigating an accident because the system will remain capable of being aligned for Emergency Core Cooling System operation in a timely manner. 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 37 continue to be met and the plant Technical Specifications will continue to provide the basis for safe plant operation.

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants"

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3.0 (NUREG-1431) provides format and content guidance for Technical Specifications for plants with Westinghouse Nuclear Steam Supply Systems and has been approved for use by the Nuclear Regulatory Commission.

NUREG-1431 does not specifically provide Technical Specification guidance for considering the safety injection system operable while incapable of automatic initiation. However, it does provide guidance for similar treatment of the residual heat removal system in a Limiting Condition for Operation Note in Technical Specification 3.5.3, "ECCS – Shutdown". The residual heat removal system is considered operable although it is not aligned for ECCS operation providing it is capable of being manually aligned for ECCS operation. Likewise, this license amendment proposes to add a Limiting Condition for Operation Note in Technical Specification 3.5.3, "ECCS – Shutdown" which will consider the safety injection system operable with the requirements of the Limiting Condition for Operation statement met, when it is incapable of automatic initiation, provided the safety injection system is capable of being manually aligned for ECCS injection. Thus, NUREG-1431 provides precedence for the type of Technical Specification requirements proposed by this license amendment request.
Thus, with the changes proposed in this license amendment request, the guidance NUREG-1431 is met as discussed above and the plant Technical Specifications will continue to provide the basis for safe plant operation.

Regulatory Requirements/Criteria 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.

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.

7.0 REFERENCES

1. Letter from the NRC to Mr. Roger O. Anderson, Technical Specification Interpretations for Auxiliary Feedwater and Safety Injection Systems Operability – Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (TAC Nos. M99666, M99667, M99669 and M99670), dated October 16, 1997.

Exhibit B

Proposed Technical Specification and Bases Changes (markup)

Technical Specification Page

3.5.3-1

Bases page
(for information only)

B 3.5.3-2

2 pages follow

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS – Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

----------NOTES----------
1. An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.
2. An SI train may be considered OPERABLE when capable of being manually aligned for ECCS injection.

APPLICABILITY: MODE 4 when both RCS cold leg temperatures are > SI pump disable temperature specified in PTLR.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to ECCS safety injection (SI) subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS safety injection (SI) subsystem inoperable.	B.1 Restore required ECCS SI subsystem to OPERABLE status.	1 hour

BASES (continued)

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the cold leg injection nozzles and reactor vessel upper plenum. In the long term, this flow path may be switched to take its supply from the containment sump.

This LCO is modified by two~~a~~ Notes. Note 1 ~~that~~ allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

Note 2 allows an SI train to be considered OPERABLE when it has been rendered incapable of injecting into the RCS due to a single action or automatic signal, if capable of being manually realigned from the control room for ECCS injection. This precludes inadvertent SI injection with the RCS in a water solid condition during MODE 4 while ensuring adequate injection capability for loss of RCS inventory events.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F and both RCS cold leg temperatures above the SI pump disable temperature specified in

Exhibit C

Proposed Technical Specification Changes (retyped)

Technical Specification Page

3.5.3-1

1 page follows

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS – Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

----------NOTES----------
1. An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.
2. An SI train may be considered OPERABLE when capable of being manually aligned for ECCS injection.

APPLICABILITY: MODE 4 when both RCS cold leg temperatures are > SI pump disable temperature specified in PTLR.

ACTIONS

----------NOTE----------
LCO 3.0.4.b is not applicable to ECCS safety injection (SI) subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS safety injection (SI) subsystem inoperable.	B.1 Restore required ECCS SI subsystem to OPERABLE status.	1 hour