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Northern States Power Company

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November 19, 1999

10 CFR Part 50
Section 50.90

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

License Amendment Request dated November 19, 1999
Establish Required Actions For Operation In Mode 3
With No RC Pumps In Operation

Attached is a request for a change to the Technical Specifications, Appendix A of the Operating Licenses, for the Prairie Island Nuclear Generating Plant (PINGP). Northern States Power Company submits this request in accordance with the provisions of 10CFR50.90.

PINGP has recently reconsidered the impact of both actual previous and postulated future disruptions of offsite power. Previous disruptions include the June 29, 1996, event (LER 1-96-12), where severe weather with high straight-line winds toppled transmission towers interrupting service on three of four 345 Kv transmission lines into the Prairie Island switchyard and the January 5, 1999, event (LER 1-99-01), where an internal electrical fault in the Unit 1 Main Auxiliary (1M) transformer and the resulting fire locked out both 1M and 1R (Unit 1 Reserve Auxiliary Transformer). Postulated potential transmission system disruptions include local weather, Y2K, and solar activity induced geomagnetic storms.

Consideration of the impact on PINGP operations resulting from disruptions of offsite power has identified two desirable changes to current Technical Specifications to enhance plant safety:
(1) With no Reactor Coolant Pumps (RCPs) in operation current Technical Specifications do not provide for any specific required action and therefore the required actions of TS 3.0.C are applicable. During a loss of offsite power this

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required action would reduce the available options for decay heat removal. This amendment request proposes to establish required actions which would permit continued operation in Mode 3 with no RCPs in operation.
(2) Current Technical Specifications permit both RCPs to be turned off for 1 hour. While this is sufficient time to conduct some testing, it is not sufficient time to conduct either preplanned maintenance or electrical lineup switching. After equipment repairs restoration of the station electrical lineup to the normal configuration used to power the RCPs currently requires performance of an unnecessary RCS cooldown and heatup transient. This amendment request proposes to permit intentional continued operation in Mode 3 with no RCPs in operation for a period not to exceed 12 hours.

Exhibit A contains a description of the proposed change, the reasons for requesting the change, the supporting safety evaluation, and the significant hazards determination. Exhibit B contains current Prairie Island Technical Specification pages marked up to show the proposed change. Exhibit C contains the revised Prairie Island Technical Specification pages incorporating the proposed change.

If you have any questions related to this license amendment request, please contact John Stanton at 651-388-1121 x4083.

Joel P. Sorensen

Joel P. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant

Attachments:

Affidavit

Exhibit A, Evaluation of Proposed Changes to the Technical Specifications Appendix A of Operating Licenses DPR-42 and DPR-60.

Exhibit B, Marked Up Technical Specification Pages

Exhibit C, Revised Technical Specification Pages

c: Regional Administrator -- III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
Steve Minn, State of Minnesota
J E Silberg

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT DOCKET Nos. 50-282
50-306

REQUEST FOR AMENDMENT TO
OPERATING LICENSES DPR-42 & DPR-60

License Amendment Request dated November 19, 1999
Establish Required Actions For Operation In Mode 3
With No RC Pumps In Operation

Northern States Power Company, a Minnesota corporation, with this letter is submitting information to support a requested license amendment. This letter and its attachments contain no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By Joel P. Sorensen

Joel P. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant

On this 19th day of November, 1999 before me a notary public in and for said County, personally appeared, Joel P. Sorensen, Site General Manager, Prairie Island Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Judith L. Rausch

JUDITH L. RAUSCH
NOTARY PUBLIC - MINNESOTA
My Commission Expires 1/31/2005

EXHIBIT A

PRAIRIE ISLAND NUCLEAR GENERATING STATION

License Amendment Request dated November 19, 1999

Evaluation of Proposed Changes to the
Technical Specification Appendix A of
Operation License DPR-42 and DPR-60

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose the following changes to the Technical Specifications contained in Appendix A of the Facility Operating Licenses:

PROPOSED CHANGE AND REASONS FOR CHANGE

TS 3.1.A.1.b Reactor Coolant System Average Temperature Above 350°F

i) Add the condition TS 3.1.A.1.b(3) which will permit both reactor coolant pumps (RCPs) to not be in operation for 72 hours provided that four specific actions are taken and that if this time limit is exceeded reactor coolant system (RCS) average temperature is reduced below 350°F within the next 12 hours. These actions are:
   (a) to immediately de-energize all control rod drive mechanisms,
   (b) to immediately suspend all operations involving a reduction of RCS boron concentration,
   (c) to maintain core outlet temperature at least 10°F below saturation temperature, and
   (d) to immediately initiate action to restore one RCS loop to OPERABLE status and operation.

ii) Identify that entry into TS 3.1.A.1.b(3) supercedes TS 3.1.A.1.b(2).

iii) The asterisk note for TS 3.1.A.1.b(1), which allows both reactor coolant pumps to be shutdown for up to 1 hour, is moved to apply to action (d) in TS 3.1.A.1.b(3) and the required actions which have been incorporated into TS 3.1.A.1.b(3) from the asterisk note are dropped from the asterisk note.

iv) The 1 hour time limit in the asterisk note for TS 3.1.A.1.b(3)(d), that allows both reactor coolant pumps to be shutdown for up to 1 hour, is increased to 12 hours. The note is modified to clarify that it applies to situations where the RCP inoperability is the result of preplanned work activities.

Basis 3.1.A Reactor Coolant System Operational Components

The bases for Specification 3.1.A.1.b is revised in accordance with the changes made in the specification as stated above. The changes to the bases are shown in Exhibit B.

After a loss of offsite power or other disruption of reactor coolant pump operation the proposed change (i) will allow the plant to remain in Mode 3 for 72 hours while efforts are undertaken to restore an RCP to operating status. This condition provides the opportunity to avoid thermal cycling of the reactor coolant (RC) and residual heat removal (RHR) systems. It also maintains the immediately available means for removing reactor core decay heat. Changing from Mode 3 to Mode 4 requires approximately 12 hours: 2 hours to establish the reactor coolant system (RCS) boron concentration needed to maintain the shutdown margin required for Mode 4 and at least another 8 hours to cooldown from 547°F to <350°F with natural circulation which is restricted to a maximum cooldown rate of 25°F/hr to preclude void formation in the reactor head.

The proposed change (ii) is necessary to avoid confusion over which required completion time is intended to be applicable when the cooldown from Mode 3 to Mode 4 must be performed using natural circulation. While the condition described in TS 3.1.A.1.b(2), one RCS loop inoperable, is true if neither RCP is in operation, the completion time limit of 6 hours to transition from Mode 3 to Mode 4 is not appropriate to plant conditions when neither RCP is operating. If at least one RCP is in operation then the RCS cooldown rate is only restricted to 100°F/hr and a completion time limit of 6 hours to transition from Mode 3 to Mode 4 is not inappropriate.

The proposed change (iii) consolidates the allowance to intentionally remove both RCPs from operation as a subset of the condition that both RCPs are currently not in operation. The proposed change (iii) provides that when the RCPs have been deliberately shutdown then the required action in TS 3.1.A.1.b(3)(d), to immediately take action to restore one RCP to operation, may be suspended for a time period such that at the end of 12 hours an RCP has been restarted and is back in operation.

Currently both RCPs may intentionally be removed from operation for a time period of 1 hour. The proposed change (iv) increases this time period to 12 hours and clarifies that it applies to situations where the RCP inoperability is the result of preplanned work activities. Prairie Island experiences, such as the January 5, 1999, internal electrical fault in the 1M Station Auxiliary transformer (LER 1-99-01), have demonstrated that it would be advantageous to have the ability to change the station electrical lineup feeding the RCPs, but this can not be accomplished in 1 hour. While the Prairie Island station electrical distribution system provides the flexibility to power any RCP from diverse power sources, such as the 1MY feed off the 1M Station Auxiliary transformer, the 1RX feed off the 1R

Reserve Auxiliary transformer, the 2RX Reserve Auxiliary transformer, and the 2MX feed off the 2M Station Auxiliary transformer, the current Technical Specifications do not provide sufficient time to accomplish all the necessary load shedding, switching, protective relay setpoint changes, and load restorations. Instead it has been required to take the affected unit to Mode 4, which puts the plant through an unnecessary cooldown and heatup transient. The safety significance of the conditions attendant with a planned work evolution that disables both RCPs for 12 hours are bounded by the safety significance of the conditions attendant with unplanned events that disable both RCPs for 72 hours. Situations that would require shutting down and disabling both RCPs while in Mode 3 are anticipated to be very infrequent. Based on past experiences this frequency is expected to average less than once per 5 years of reactor operation.

SAFETY EVALUATION

Recent consideration of postulated Y2K induced transmission system disruptions has identified that, while the expected probability of any such disruptions are small, the possibility does exist. It is also expected that were a loss of offsite power to occur at Prairie Island as a result of transmission system disruptions, the availability of offsite power would be restored to satisfy the conditions and time limit in TS 3.7.B.4, though it is not expected that an RCP could be restarted within the time limit of TS 3.0.C. The restoration performance demonstrated after the severe weather physical damage to transmission lines that occurred on June 29, 1996, suggests that a Y2K transmission system disruption could be fixed and an RCP restarted within the 72 hour time limit of TS 3.1.A.1.b(3).

Allowing the plant to remain in Mode 3 for 72 hours and remove decay heat by means of natural circulation, operation of the turbine driven auxiliary feedwater pump and operation of either the steam generator power operated relief valves or safety relief valve provides several desirable plant safety enhancements:

(1) If offsite power is restored and the RCPs are restarted within the 72 hours, thermal cycling of the RC and RHR systems has been avoided.

(2) If offsite power is restored and the RCPs are restarted within the 72 hours, the need for operations staff to perform a natural circulation cooldown either with or without the "non-vital AC powered" control rod drive mechanism (CRDM) cooling fans has been avoided. After a reactor trip it is preferable to have the option to establish and maintain a steady state condition for a time rather than to be required to proceed swiftly into a cooldown transient.

(3) The availability of a method for removing decay heat that is not dependent on diesel generator performance has been preserved.

(4) After 72 hours the decay heat rate will be less than half of the decay heat rate at the 6 hour mark. If offsite power has not been restored and the RCPs have not been restarted within the 72 hours, the reactor coolant system average temperature must be reduced below 350°F within the next 12 hours. Although the steam turbine driven auxiliary feedwater (TDAFW) pump will be operating below nominal design capacity when the RCS average temperature is below 350°F, significant margin exists between the heat removal capacity provided by the available TDAFW flow and the core decay heat rate 72 hours after the reactor trip.

(5) If offsite power is lost to both units and sufficient offsite power will not be available to permit an RCP on each or either unit to be restarted within the 72 hours, the two units can first be stabilized in natural circulation operation and then taken sequentially to Mode 4. When handling a transient on a unit, it is preferable to have the other unit maintaining a stable condition.

NSPNAD-8102-P, Revision 7, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods For Application to PI Units" dated January, 1999, and submitted to the NRC for review and approval on January 29, 1999, describes a new analysis methodology to determine shutdown margin requirements for Modes 3, 4, 5, and 6, based on ensuring that a complete loss of shutdown margin will not occur for at least 24 minutes from the initiation of dilution in an uncontrolled boron dilution accident. This analysis methodology utilizes the assumption that "the boron concentration is uniform throughout the mass being diluted, i.e. perfect and instantaneous mixing." The analysis methodology does not address the situation where a slug of dilute fluid enters a core region of high excess reactivity and creates the potential for a momentary localized critical condition. Because natural circulation is not capable of ensuring a sufficiently rapid and uniform mixing of fluid in the RCS as assumed in this analysis methodology, the requirement that boron dilution operations must be immediately suspended with neither reactor coolant pump in operation is utilized to provide assurance that a momentary localized critical condition will not occur. During operation in Mode 3 it will be necessary to increase boron concentration to establish conditions providing for the required shutdown margin identified in the analysis of the uncontrolled boron dilution accident before the negative reactivity from xenon peaks and decays toward zero power equilibrium levels. Because this is a rather slow transient, the degree of RCS fluid mixing provided by natural circulation is quite adequate[1] for increasing the boron concentration in the RCS. Increasing the time the plant is allowed to remain in Mode 3 from the 6 hours currently specified under TS 3.0.C to the 72 hours specified in this proposed change will not impact the currently available ability to establish and maintain the required shutdown margin using natural circulation.

Natural circulation flow versus reactor power has been calculated by Westinghouse using an analytical model based on the conditions of equilibrium flow and maximum loop flow impedance. The results from this model have been validated against measured data for a variety of Westinghouse PWRs such as Yankee-Rowe, San Onofre, Connecticut Yankee, and Ginna. The calculated natural circulation flow for Ginna was within 5% of the values measured during a January 18, 1970 test at Ginna. The calculated natural circulation flow values[2] presented in the Point Beach and Ginna UFSARs are in close agreement to each

[1] WCAP 11095 describes a natural circulation, boron mixing and cooldown test conducted at Diablo Canyon on March 29, 1985, which provided information on the time delay associated with boron mixing under natural circulation conditions. A charging pump was aligned to the boron injection tank for 20 minutes at a flow rate of 150 gpm to flush a total of 900 gallons of 21000 ppm borated water into the RCS. Approximately 12 minutes after this boron injection was completed the RCS boron concentration increase was 280 ppm and within the hour the measured increase had settled at approximately 300 ppm. This increase in boron concentration was judged to be sufficiently quick to ensure the rapid and adequate mixing of boron added to the RCS under natural circulation conditions.

[2]

% Rated Thermal Power	Calculated Point Beach % Rated Flow	Calculated Ginna % Rated Flow	Measured Ginna % Rated Flow
5.0	-	5.5	-
4.2	-	5.1	5.25
3.5	4.6	4.7	-
3.0	4.3	4.5	-
2.5	4.0	4.1	-
2.0	3.7	3.8	4.0
1.5	3.3	3.4	-
1.0	2.9	3.0	-
0.5	-	2.4	-

other and demonstrate that the flow to power ratio for natural circulation is always greater than 1.0 and that the ratio increases as the decay heat power levels decreases. Because Prairie Island, Point Beach and Ginna have nearly identical reactor coolant system designs, the natural circulation conditions calculated for Point Beach and Ginna support the conclusion that extended natural circulation operation at Prairie Island will not degrade reactor fuel thermal margins.

To maintain a unit in Mode 3 under natural circulation a supply of cooling water for the auxiliary feedwater system to pump to the steam generators must be available that is both sufficient to remove decay heat during the period of steady state natural circulation operation in Mode 3 and sufficient to remove decay heat and the latent heat in reactor coolant system during cooldown to Mode 4. An ample quantity of cooling water can be supplied by the cooling water system drawing suction on the river. For economic reasons related to steam generator performance and steam generator secondary side chemistry control the preferred source of cooling water is condensate water. Condensate water is available from three 150,000 gallon capacity condensate storage tanks. The quantity of water in these tanks is sufficient to support natural circulation operation for significant periods of time.

A current basis for Prairie Island Technical Specifications states that two methods of removing heat are required at all times except during refueling and that above 350°F both reactor coolant loops must be operable to serve this function. Technical Specification TS 3.1.A.1.b.(1) indicates that for a reactor coolant loop to be operable the associated steam generator and reactor coolant pump must be operable. Operation under natural circulation conditions is not discussed in the current Technical Specification basis. While the flow rate produced in a reactor coolant loop under natural circulation conditions is much less than that produced by an operating reactor coolant pump, the natural circulation flow rate in Mode 3 is sufficient, as discussed previously, to remove decay heat from the core and deliver it to a steam generator at a rate that maintains adequate thermal margins in the core. Buoyancy forces generated by water density differences and elevation differences in the reactor coolant system will establish and maintain the natural circulation flow so long as significant void formation in the reactor coolant system is prevented and steam generator water level is maintained. Void formation is precluded by the required action in the proposed change to maintain core outlet temperature at least 10°F below saturation temperature. Steam generator level is maintained by flow from the auxiliary feedwater system, which has both a steam turbine driven pump and a motor driven pump powered from a vital electrical power supply, that are each capable of supplying 100% of the required design cooling water flow rate to both steam generators. Natural circulation flow in the RCS in conjunction with auxiliary feedwater flow to a steam generator will be heated and discharged to atmosphere by either the steam generator's power operated relief valve or its safety relief valve. It has been judged that a single operable steam generator under natural circulation conditions is sufficient to remove decay heat, and that two operable steam generators with natural circulation operation of the reactor coolant system provides the redundancy of safety function intended by the current Technical Specification basis.

This proposed change is consistent with earlier Prairie Island Technical Specifications (Rev 45 (4-1-81)), which provided a 72 hour period with RCS temperature above 350°F to restore a reactor coolant pump to an operable status before requiring a reduction in RCS average temperature to below 350°F. Except for the extension of the time allowed to deliberately have both RCPs shutdown/inoperable, this proposed change is also consistent with NUREG 1431 Revision 1, Westinghouse Standard Technical Specifications, LCO 3.4.5 ACTION D.

DETERMINATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not significantly affect any system that is a contributor to initiating events for previously evaluated accidents. The probability of occurrence for the "Uncontrolled RCCA Withdrawal From a Subcritical Condition" abnormal operational transient will be decreased by the actions required by the proposed change, and the consequences will remain unchanged. The probability of occurrence and consequences for the "Chemical and Volume Control System Malfunction" (Uncontrolled Boron Dilution) abnormal operational transient will not be changed by the actions required by the proposed change. Neither does the change significantly affect any system that is used to mitigate any previously evaluated accidents. The proposed change extends the time that the plant can remain in Mode 3 on natural circulation. This will not degrade the ability of the plant to later reduce reactor coolant system temperature and pressure to Mode 4 conditions where the diesel generators and RHR system are still available to remove decay heat. The proposed changes do not involve any significant increase in the probability or consequence of an accident previously evaluated.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

   The proposed change does not alter the design, function, or manner of operation of any plant component and does not install any new or different equipment. The proposed change extends the time that the plant can remain in Mode 3 on natural circulation. A possibility of a new or different kind of accident from those previously analyzed has not been created.

3. <u>The proposed amendment will not involve a significant reduction in a margin of safety.</u>

   The proposed change extends the time that the plant can remain in Mode 3 on natural circulation. Sufficient capacity to remove decay heat is still available. Under natural circulation conditions the availability of both steam generators provides the expected redundancy of this required safety function associated with the reactor coolant system Technical Specification basis. The proposed change does not involve a significant reduction in the margin of safety associated with the safety limits inherent in either the principle barriers to a radiation release (fuel cladding, RCS boundary, and reactor containment), the maintenance of critical safety functions (subcriticality, core cooling, ultimate heat sink, RCS inventory, RCS boundary integrity, and containment integrity), or other structures, systems or components (SSCs) significant to safety.

Considering the above evaluation and pursuant to 10CFR50.91, Northern States Power Company has determined that operation of the Prairie Island Nuclear Generating Plant in accordance with the proposed license amendment request does not involve a significant hazards consideration as defined by Nuclear Regulatory Commission regulations in 10CFR50.92.

ENVIRONMENTAL ASSESSMENT

Northern States Power Company has evaluated the proposed change and determined that:

1. The change does not involve a significant hazards consideration,
2. The change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and
3. The change does not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), an environmental assessment of the proposed changes is not required.

EXHIBIT B

PRAIRIE ISLAND NUCLEAR GENERATING STATION

License Amendment Request dated November 19, 1999

Appendix A, Technical Specification Pages

Marked Up Pages
(shaded material to be added, strikethrough material to be removed)

TS.3.1-1

B.3.1-1

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system when irradiated fuel is in the containment.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to assure safe reactor operation.

Specification

A. Operational Components

1. Reactor Coolant Loops and Coolant Circulation

a. Reactor Critical

(1) A reactor shall not be made or maintained critical unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are in operation, except 1) during low power PHYSICS TESTS or 2) as specified in 3.1.A.1.a.(2) below.

(2) With less than the above required reactor coolant loops in operation, be in at least HOT SHUTDOWN within 6 hours.

b. Reactor Coolant System Average Temperature Above 350°F.

(1) Reactor coolant system average temperature shall not exceed 350°F unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are OPERABLE with at least one reactor coolant loop in operation* (except as specified in 3.1.A.1.b(2) and 3.1.A.1.b(3) below).

(2) A reactor coolant loop may be inoperable for 72 hours provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, reduce reactor coolant system average temperature below 350°F within the next 6 hours.

(3) With both reactor coolant pumps inoperable or not in operation immediately:
(a) De-energize all control rod drive mechanisms,
(b) Suspend all operations involving a reduction of RCS boron concentration,
(c) Establish and maintain the core outlet temperature at least 10°F below saturation temperature, and
(d) Initiate action to restore one reactor coolant pump to OPERABLE status and operation.*

If at least one reactor coolant pump is not restored to OPERABILITY and operation within 72 hours, reduce reactor coolant system average temperature to below 350°F within the next 12 hours. While applicable, this specification supercedes 3.1.A.1.b(2).

* If the RCP shutdown or inoperability was due to preplanned work activities, such as testing, switching, or maintenance, immediate restoration action is not required, but if at least one reactor coolant pump is not restored to operability and operation within 12 hours, reduce reactor coolant system average temperature to below 350°F within the next 12 hours.
~~Both pumps may be shutdown for up to one hours provided the reactor is subcritical, the reactor trip breakers are open, no operations are permitted that would cause dilution of the reactor coolant boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.~~

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth greater than or equal to 50% of the 0.050-inch tube wall thickness as being unacceptable for POWER OPERATION. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents (Reference 2).

Part A of the specification requires that both reactor coolant loops be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30 for Exxon fuel and 1.17 for Westinghouse fuel. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power PHYSICS TESTS, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

Two methods of removing decay heat are required at all times except during REFUELING. Above 350°F, having both reactor coolant loops ~~must be~~ OPERABLE (each associated reactor coolant pump and steam generator is OPERABLE) ~~to serve~~ will provide the required redundancy of this safety function. In Mode 3 with natural circulation in the reactor coolant system the required redundancy of the safety function, removal of decay heat, will be provided by having both steam generators OPERABLE. Below 350°F an OPERABLE reactor coolant loop or an OPERABLE residual heat removal loop is capable of removing decay heat and any combination of two OPERABLE loops ~~serve~~ provide ~~this~~ the required redundancy of this safety function.

Specification 3.1.A.1.d.(2) allows the use of one safety injection pump to ensure that adequate core cooling and reactor coolant system inventory can be maintained in the event of a loss of Residual Heat Removal System cooling during reduced inventory conditions. A reduced inventory condition, as defined by Generic Letter 88-17, Loss of Decay Heat Removal, exists whenever the reactor vessel water level is lower than three feet below the reactor vessel flange. The operation of a safety injection pump under such conditions would be controlled by an approved emergency operating procedure.

EXHIBIT C

PRAIRIE ISLAND NUCLEAR GENERATING STATION

License Amendment Request dated November 19, 1999

Appendix A, Technical Specification Pages

Revised Pages

TS.3.1-1

B.3.1-1

3.1 REACTOR COOLANT SYSTEM

<u>Applicability</u>

Applies to the operating status of the reactor coolant system when irradiated fuel is in the containment.

<u>Objective</u>

To specify those limiting conditions for operation of the reactor coolant system which must be met to assure safe reactor operation.

<u>Specification</u>

A. <u>Operational Components</u>

1. <u>Reactor Coolant Loops and Coolant Circulation</u>

a. <u>Reactor Critical</u>

(1) A reactor shall not be made or maintained critical unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are in operation, except 1) during low power PHYSICS TESTS or 2) as specified in 3.1.A.1.a.(2) below.

(2) With less than the above required reactor coolant loops in operation, be in at least HOT SHUTDOWN within 6 hours.

b. <u>Reactor Coolant System Average Temperature Above 350°F.</u>

(1) Reactor coolant system average temperature shall not exceed 350°F unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are OPERABLE with at least one reactor coolant loop in operation (except as specified in 3.1.A.1.b(2) and 3.1.A.1.b(3) below).

(2) A reactor coolant loop may be inoperable for 72 hours provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If OPERABILITY is not restored within the time specified, reduce reactor coolant system average temperature below 350°F within the next 6 hours.

(3) With both reactor coolant pumps inoperable or not in operation immediately:
(a) De-energize all control rod drive mechanisms,
(b) Suspend all operations involving a reduction of RCS boron concentration,
(c) Establish and maintain core outlet temperature at least 10°F below saturation temperature, and
(c) Initiate action to restore one reactor coolant pump to OPERABLE status and operation.*

If at least one reactor coolant pump is not restored to OPERABILITY and operation within 72 hours, reduce reactor coolant system average temperature to below 350°F within the next 12 hours. While applicable, this specification supercedes 3.1.A.1.b(2).

* If the RCP shutdown or inoperability was due to preplanned work activities, such as testing, switching, or maintenance, immediate restoration action is not required, but if at least one reactor coolant pump is not restored to OPERABILITY and operation within 12 hours, reduce reactor coolant system average temperature to below 350°F within the next 12 hours.

3.1 REACTOR COOLANT SYSTEM

Bases continued

A. Operational Components

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth greater than or equal to 50% of the 0.050-inch tube wall thickness as being unacceptable for POWER OPERATION. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents (Reference 2).

Part A of the specification requires that both reactor coolant loops be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30 for Exxon fuel and 1.17 for Westinghouse fuel. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power PHYSICS TESTS, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

Two methods of removing decay heat are required at all times except during REFUELING. Above 350°F, having both reactor coolant loops OPERABLE (each associated reactor coolant pump and steam generator is OPERABLE) will provide the required redundancy of this safety function. In Mode 3 with natural circulation in the reactor coolant system the required redundancy of the safety function, removal of decay heat, will be provided by having both steam generators OPERABLE. Below 350°F an OPERABLE reactor coolant loop or an OPERABLE residual heat removal loop is capable of removing decay heat and any combination of two OPERABLE loops provide the required redundancy of this safety function.

Specification 3.1.A.1.d.(2) allows the use of one safety injection pump to ensure that adequate core cooling and reactor coolant system inventory can be maintained in the event of a loss of Residual Heat Removal System cooling during reduced inventory conditions. A reduced inventory condition, as defined by Generic Letter 88-17, Loss of Decay Heat Removal, exists whenever the reactor vessel water level is lower than three feet below the reactor vessel flange. The operation of a safety injection pump under such conditions would be controlled by an approved emergency operating procedure.