Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

March 1999

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: BORIC ACID STORAGE TANK LEVEL INSTRUMENTATION (TAC NOS. MA4228 AND MA4229)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 143 to Facility Operating License No. DPR-42 and Amendment No. 134 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 25, 1998. The amendments revise TS 3.2 and Table 3.5-2B to allow limited inoperability of boric acid storage tank level channels and transfer logic channels to provide for required testing and maintenance of the associated components.

The NRC staff met with representatives from PINGP at NRC Headquarters on January 7, 1999, concerning your amendment request. The PINGP staff requested the meeting to brief the NRC staff on the background of the amendment request and also to step through the relevant logic system diagrams with the NRC staff. The NRC staff found the meeting very helpful in understanding the relatively complex logic systems involved in the license amendment request as well as PINGP's need for the amendment, and therefore preempting the need for a staff request for additional information. As a result, the staff was able to accommodate PINGP's request for an expedited review schedule.

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

ORIGINAL SIGNED BY

Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 143 to DPR-42
2. Amendment No. 134 to DPR-60
3. Safety Evaluation

cc w/encl: See next page
DISTRIBUTION: See attached page
DOCUMENT NAME: G:\WPDOCS\PRAIRIE\AMD4228.WPD *No major changes to SE
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Mr. Roger O. Anderson, Director
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Prairie Island Nuclear Generating Plant

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Tribal Council
Prairie Island Indian Community
ATTN: Environmental Department
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Welch, Minnesota 55089

DATED: March 17, 1999

AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File (50-282, 50-306)
PUBLIC
PDIII-1 Reading
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G. Hill (4)
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 143
License No. DPR-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated November 25, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-42 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 143 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Tae Kim

Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 17, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS.3.2-1	TS.3.2-1
Table TS.3.5-2B (Page 6 of 9)	Table TS.3.5-2B (Page 6 of 9)
Table TS.3.5-2B (Page 9 of 9)	Table TS.3.5-2B (Page 9 of 9)
B.3.5-1	B.3.5-1
B.3.5-4	B.3.5-4
B.3.5-5	B.3.5-5
- -	B.3.5-6

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

<u>Applicability</u>

Applies to the operational status of the chemical and volume control system.

<u>Objective</u>

To define those conditions of the chemical and volume control system necessary to assure safe reactor operation and safe COLD SHUTDOWN.

<u>Specification</u>

A. When fuel is in a reactor and reactor coolant system average temperature is at or below 200°F there shall be at least one flow path to the core for boric acid injection. If no OPERABLE flow path exists, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

B. A reactor shall not be made or maintained critical nor shall the reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.2.C or 3.2.D below, or Table TS.3.5-2B):

1. Two of the three charging pumps shall be OPERABLE.

2. At least one boric acid tank shall be aligned to the unit and shall contain a minimum of 2000 gallons of 11.5% to 13% by weight boric acid solution at a temperature of at least 145°F.

3. System piping, valves and pumps shall be OPERABLE to the extent of establishing two independent flow paths for boric acid injection -- one flow path from the boric acid tanks to the core and one flow path from the refueling water storage tank to the core.

4. Two channels of heat tracing shall be OPERABLE for the flow paths from the boric acid tanks required to meet the requirements of Specification 3.2.B.3.

5. Automatic valves, piping, and interlocks associated with the above components which are required to operate for the steam line break accident are OPERABLE.

TABLE TS.3.5-2B (Page 6 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. LOSS OF POWER					
a. Degraded Voltage 4kV Safeguards Bus	4/Bus (2/phase on 2 phases)	2/Bus (1/phase on 2 phases)	3/Bus	1, 2, 3, 4	31, 32, 33
b. Undervoltage 4kV Safeguards Bus	4/Bus (2/phase on 2 phases)	2/Bus (1/phase on 2 phases)	3/Bus	1, 2, 3, 4	31, 32, 33
9. BORIC ACID STORAGE TANK					
a. Lo-Lo Level	2 channels with 2 sensors per channel	1 sensor per channel in both channels	2 sensors in one channel	1, 2, 3, 4	34
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	35, 36

TABLE 3.5-2B (Page 9 of 9)

Action Statements

ACTION 30: With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.4.2. However, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 31: With the number of OPERABLE channels one less than the Total Number of Channels, operation in the applicable MODE may proceed provided the inoperable channel is placed in the bypassed condition within 6 hours.

ACTION 32: With the number of OPERABLE channels two less than the Total Number of Channels, operation in the applicable MODE may proceed provided the following conditions are satisfied:

a. One inoperable channel is placed in the bypassed condition within 6 hours, and,

b. The other inoperable channel is placed in the tripped condition within 6 hours, and,

c. All of the channels associated with the redundant 4kV Safeguards Bus are OPERABLE.

ACTION 33: If the requirements of ACTIONS 30 or 31 cannot be met within the time specified, or with the number of OPERABLE channels three less than the Total Number of Channels, declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.7.B.

ACTION 34: With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 6 hours and the Minimum Channels OPERABLE requirement is met. Restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 35: With one channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 36: Two channels may be inoperable for up to 1 hour for surveillance testing per Specification 4.1. Restore at least one channel to OPERABLE status within this 1 hour or initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.5 INSTRUMENTATION SYSTEM

Bases

Instrumentation has been provided to sense accident conditions and to initiate reactor trip and operation of the Engineered Safety Features (Reference 1). The OPERABILITY of the Reactor Trip System and the Engineered Safety System instrumentation and interlocks ensures that: (1) the associated ACTION and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, (3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analysis.

Specified surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. (Boric Acid Storage Tank instrumentation which provides automatic transfer of safety injection suction was not modeled in this WCAP.)

The evaluation of surveillance frequencies and out of service times for the reactor protection and engineered safety feature instrumentation described in WCAP-10271 included the allowance for testing in bypass. The evaluation assumed that the average amount of time the channels within a given trip function would be in bypass for testing was 4 hours.

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss-of-coolant accident (LOCA) is provided by a high containment pressure signal backed up by the low pressurizer pressure signal. These conditions would accompany the depressurization and coolant loss during a LOCA.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the low pressurizer pressure signal and, in case of a break within the containment, by the high containment pressure signal.

The safety injection of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Automatic Transfer of Safety Injection Suction

The plant is equipped with three boric acid storage tanks for the two units. One tank is normally aligned to the safety injection system for each unit. Following initiation of the Engineered Safety Features, the safety injection pumps take suction from the aligned boric acid storage tank. When the boric acid storage tank level falls to the lo-lo level, an interlock automatically transfers the safety injection pumps suction from the boric acid storage tank to the refueling water storage tank. The boric acid storage tank that is not aligned to either unit, including its associated piping and interlocks, is not required to be OPERABLE.

Limiting Instrument Setpoints

1. The high containment pressure limit is set at about 10% of the maximum internal pressure. Initiation of Safety Injection protects against loss of coolant (Reference 2) or steam line break accidents as discussed in the safety analysis.

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant (Reference 2) or steam line break accidents (Reference 3) as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis (Reference 2).

4. The steam line low pressure signal is lead/lag compensated and its set-point is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis (Reference 3).

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full-load flow at the full load pressure in order to protect against large steam break accidents. The coincident low T_{avg} setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam break (Reference 3).

6. Steam generator low-low water level and 4.16 kV Bus 11 and 12 (21 and 22 in Unit 2) low bus voltage provide initiation signals for the Auxiliary Feedwater System. Selection of these setpoints is discussed in the Bases of Section 2.3 of the Technical Specification.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Limiting Instrument Setpoints (continued)

7. High radiation signals providing input to the Containment Ventilation Isolation circuitry are set in accordance with the Radioactive Effluent Technical Specifications. The setpoints are established to prevent exceeding the limits of 10 CFR Part 20 at the SITE BOUNDARY.

8. The degraded voltage protection setpoint is ≥94.8% and ≤96.2% of nominal 4160 V bus voltage. Testing and analysis have shown that all safeguards loads will operate properly at or above the minimum degraded voltage setpoint. The maximum degraded voltage setpoint is chosen to prevent unnecessary actuation of the voltage restoring scheme at the minimum expected grid voltage. The first degraded voltage time delay of 8 ± 0.5 seconds has been shown by testing and analysis to be long enough to allow for normal transients (i.e., motor starting and fault clearing). It is also longer than the time required to start the safety injection pump at minimum voltage. The second degraded voltage time delay is provided to allow the degraded voltage condition to be corrected within a time frame which will not cause damage to permanently connected Class 1E loads.

   The undervoltage setpoint is 75 ± 2.5% of nominal bus voltage. The minimum setpoint ensures equipment operates above the limiting value of 75% (of 4000 V) for one minute operation. The 75% maximum setpoint is chosen to prevent unnecessary actuation of the voltage restoring scheme during voltage dips which occur during motor starting. The undervoltage time delay of 4 ± 1.5 seconds has been shown by testing and analysis to be long enough to allow for normal transients and short enough to operate prior to the degraded voltage logic, providing a rapid transfer to an alternate source.

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for CHANNEL CALIBRATION and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not

3.5 INSTRUMENTATION SYSTEM

Bases continued

Instrument Operating Conditions (continued)

intentionally placed in a tripped mode since these are one-out of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in a concurrent channel.

References

1. USAR, Section 7.4.2
2. USAR, Section 14.6.1
3. USAR, Section 14.5.5
4. FSAR, Appendix I

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. DPR-60

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (the licensee) dated November 25, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 134 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Tae Kim

Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 17, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 134

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS.3.2-1	TS.3.2-1
Table TS.3.5-2B (Page 6 of 9)	Table TS.3.5-2B (Page 6 of 9)
Table TS.3.5-2B (Page 9 of 9)	Table TS.3.5-2B (Page 9 of 9)
B.3.5-1	B.3.5-1
B.3.5-4	B.3.5-4
B.3.5-5	B.3.5-5
- -	B.3.5-6

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the chemical and volume control system.

Objective

To define those conditions of the chemical and volume control system necessary to assure safe reactor operation and safe COLD SHUTDOWN.

Specification

A. When fuel is in a reactor and reactor coolant system average temperature is at or below 200°F there shall be at least one flow path to the core for boric acid injection. If no OPERABLE flow path exists, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

B. A reactor shall not be made or maintained critical nor shall the reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.2.C or 3.2.D below, or Table TS.3.5-2B):

1. Two of the three charging pumps shall be OPERABLE.

2. At least one boric acid tank shall be aligned to the unit and shall contain a minimum of 2000 gallons of 11.5% to 13% by weight boric acid solution at a temperature of at least 145°F.

3. System piping, valves and pumps shall be OPERABLE to the extent of establishing two independent flow paths for boric acid injection -- one flow path from the boric acid tanks to the core and one flow path from the refueling water storage tank to the core.

4. Two channels of heat tracing shall be OPERABLE for the flow paths from the boric acid tanks required to meet the requirements of Specification 3.2.B.3.

5. Automatic valves, piping, and interlocks associated with the above components which are required to operate for the steam line break accident are OPERABLE.

TABLE TS.3.5-2B (Page 6 of 9)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. LOSS OF POWER					
a. Degraded Voltage 4kV Safeguards Bus	4/Bus (2/phase on 2 phases)	2/Bus (1/phase on 2 phases)	3/Bus	1, 2, 3, 4	31, 32, 33
b. Undervoltage 4kV Safeguards Bus	4/Bus (2/phase on 2 phases)	2/Bus (1/phase on 2 phases)	3/Bus	1, 2, 3, 4	31, 32, 33
9. BORIC ACID STORAGE TANK					
a. Lo-Lo Level	2 channels with 2 sensors per channel	1 sensor per channel in both channels	2 sensors in one channel	1, 2, 3, 4	34
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	35, 36

TABLE 3.5-2B (Page 9 of 9)

Action Statements

ACTION 30: With the number of OPERABLE channels one less than the Total Number of Channels, declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.4.2. However, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

ACTION 31: With the number of OPERABLE channels one less than the Total Number of Channels, operation in the applicable MODE may proceed provided the inoperable channel is placed in the bypassed condition within 6 hours.

ACTION 32: With the number of OPERABLE channels two less than the Total Number of Channels, operation in the applicable MODE may proceed provided the following conditions are satisfied:

a. One inoperable channel is placed in the bypassed condition within 6 hours, and,

b. The other inoperable channel is placed in the tripped condition within 6 hours, and,

c. All of the channels associated with the redundant 4kV Safeguards Bus are OPERABLE.

ACTION 33: If the requirements of ACTIONS 30 or 31 cannot be met within the time specified, or with the number of OPERABLE channels three less than the Total Number of Channels, declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.7.B.

ACTION 34: With the number of OPERABLE channels less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 6 hours and the Minimum Channels OPERABLE requirement is met. Restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 35: With one channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 36: Two channels may be inoperable for up to 1 hour for surveillance testing per Specification 4.1. Restore at least one channel to OPERABLE status within this 1 hour or initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.5 INSTRUMENTATION SYSTEM

Bases

Instrumentation has been provided to sense accident conditions and to initiate reactor trip and operation of the Engineered Safety Features (Reference 1). The OPERABILITY of the Reactor Trip System and the Engineered Safety System instrumentation and interlocks ensures that: (1) the associated ACTION and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, (3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analysis.

Specified surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. (Boric Acid Storage Tank instrumentation which provides automatic transfer of safety injection suction was not modeled in this WCAP.)

The evaluation of surveillance frequencies and out of service times for the reactor protection and engineered safety feature instrumentation described in WCAP-10271 included the allowance for testing in bypass. The evaluation assumed that the average amount of time the channels within a given trip function would be in bypass for testing was 4 hours.

Safety Injection

The Safety Injection System is actuated automatically to provide emergency cooling and reduction of reactivity in the event of a loss-of-coolant accident or a steam line break accident.

Safety injection in response to a loss-of-coolant accident (LOCA) is provided by a high containment pressure signal backed up by the low pressurizer pressure signal. These conditions would accompany the depressurization and coolant loss during a LOCA.

Safety injection in response to a steam line break is provided directly by a low steam line pressure signal, backed up by the low pressurizer pressure signal and, in case of a break within the containment, by the high containment pressure signal.

The safety injection of highly borated water will offset the temperature-induced reactivity addition that could otherwise result from cooldown following a steam line break.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Automatic Transfer of Safety Injection Suction

The plant is equipped with three boric acid storage tanks for the two units. One tank is normally aligned to the safety injection system for each unit. Following initiation of the Engineered Safety Features, the safety injection pumps take suction from the aligned boric acid storage tank. When the boric acid storage tank level falls to the lo-lo level, an interlock automatically transfers the safety injection pumps suction from the boric acid storage tank to the refueling water storage tank. The boric acid storage tank that is not aligned to either unit, including its associated piping and interlocks, is not required to be OPERABLE.

Limiting Instrument Setpoints

1. The high containment pressure limit is set at about 10% of the maximum internal pressure. Initiation of Safety Injection protects against loss of coolant (Reference 2) or steam line break accidents as discussed in the safety analysis.

2. The Hi-Hi containment pressure limit is set at about 50% of the maximum internal pressure for initiation of containment spray and at about 30% for initiation of steam line isolation. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant (Reference 2) or steam line break accidents (Reference 3) as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis (Reference 2).

4. The steam line low pressure signal is lead/lag compensated and its set-point is set well above the pressure expected in the event of a large steam line break accident as shown in the safety analysis (Reference 3).

5. The high steam line flow limit is set at approximately 20% of nominal full-load flow at the no-load pressure and the high-high steam line flow limit is set at approximately 120% of nominal full-load flow at the full load pressure in order to protect against large steam break accidents. The coincident low T_{avg} setting limit for steam line isolation initiation is set below its hot shutdown value. The safety analysis shows that these settings provide protection in the event of a large steam break (Reference 3).

6. Steam generator low-low water level and 4.16 kV Bus 11 and 12 (21 and 22 in Unit 2) low bus voltage provide initiation signals for the Auxiliary Feedwater System. Selection of these setpoints is discussed in the Bases of Section 2.3 of the Technical Specification.

3.5 INSTRUMENTATION SYSTEM

Bases continued

Limiting Instrument Setpoints (continued)

7. High radiation signals providing input to the Containment Ventilation Isolation circuitry are set in accordance with the Radioactive Effluent Technical Specifications. The setpoints are established to prevent exceeding the limits of 10 CFR Part 20 at the SITE BOUNDARY.

8. The degraded voltage protection setpoint is ≥94.8% and ≤96.2% of nominal 4160 V bus voltage. Testing and analysis have shown that all safeguards loads will operate properly at or above the minimum degraded voltage setpoint. The maximum degraded voltage setpoint is chosen to prevent unnecessary actuation of the voltage restoring scheme at the minimum expected grid voltage. The first degraded voltage time delay of 8 ± 0.5 seconds has been shown by testing and analysis to be long enough to allow for normal transients (i.e., motor starting and fault clearing). It is also longer than the time required to start the safety injection pump at minimum voltage. The second degraded voltage time delay is provided to allow the degraded voltage condition to be corrected within a time frame which will not cause damage to permanently connected Class 1E loads.

   The undervoltage setpoint is 75 ± 2.5% of nominal bus voltage. The minimum setpoint ensures equipment operates above the limiting value of 75% (of 4000 V) for one minute operation. The 75% maximum setpoint is chosen to prevent unnecessary actuation of the voltage restoring scheme during voltage dips which occur during motor starting. The undervoltage time delay of 4 ± 1.5 seconds has been shown by testing and analysis to be long enough to allow for normal transients and short enough to operate prior to the degraded voltage logic, providing a rapid transfer to an alternate source.

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for CHANNEL CALIBRATION and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not

3.5 INSTRUMENTATION SYSTEM

Bases continued

Instrument Operating Conditions (continued)

intentionally placed in a tripped mode since these are one-out of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in a concurrent channel.

References

1. USAR, Section 7.4.2
2. USAR, Section 14.6.1
3. USAR, Section 14.5.5
4. FSAR, Appendix I

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 143

TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 134 TO FACILITY OPERATION LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated November 25, 1998, Northern States Power Company (NSP, the licensee) requested an amendment to the Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2 Operating Licenses that would authorize changes to the Technical Specification (TS) for the boric acid storage tank (BAST) level instrumentation and transfer logic.

The PINGP TSs currently require monthly operability testing of the BAST transfer logic associated with the transfer of safety injection pump suction from the BAST to the refueling water storage tank (RWST). To perform this required testing, individual BAST level and transfer logic channels must be made inoperable. However, the TSs do not permit inoperability of those channels for performing the required testing. The proposed TS changes would permit limited inoperability for performing these tests.

2.0 EVALUATION

The licensee proposed changes to the following TSs:

a. TS 3.2.B

The licensee proposed a revision to TS 3.2.B to reference revised Table TS.3.5-2B which contains actions required in case of inoperability of the instrumentation required to be operable by TS 3.2.B.5, for transferring safety injection pump suction from the BAST to the RWST.

b. Table TS.3.5-2B Functional Unit 9a, BAST Lo-Lo Level

The licensee proposed a revision to Table TS.3.5-2B to add Functional Unit 9a, BAST Lo-Lo Level. Functional Unit 9a provides the requirements for operability of the BAST lo-lo level function. The proposed TS total number of channels for lo-lo level operability is 2 with 2 sensors per channel. The number of sensors required to activate the BAST to RWST transfer is 1 per channel. The proposed TS minimum number of sensors required to be operable per channel is 2. The proposed TS required number of operable channels for this functional unit is consistent in format and number with other instrumentation in current Table TS.3.5-2B with similar logic configurations. The proposed TS applicable operability modes, 1, 2, 3, and 4, are the same as those specified for the transfer logic in TS 3.2.B.5.

c. Table TS.3.5-2B, Action 34

Proposed Action 34 specifies that with the number of operable channels less than the total number of channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. Placing an inoperable channel in the tripped condition ensures that if the BAST to RWST transfer becomes necessary, the inoperable channel will not prevent the transfer. The 6-hour allowance for completion of this action allows adequate time for personnel to complete tripping of the channel, while minimizing the time and associated risk that the transfer function could be defeated by an inoperable channel during an event. The proposed requirement to place an inoperable channel in the tripped condition within 6 hours is consistent with the actions specified for other similar instrumentation in current Table TS.3.5-2B.

Proposed Action 34 also specifies a 72-hour limitation on operation with an inoperable BAST level channel. This 72-hour allowance limits the risk from a premature transfer from the BAST to the RWST while providing time for testing and repair or replacement of failed components. The 72-hour allowance for operation with an inoperable BAST level channel is comparable to the allowed out-of-service time specified in TS 3.2.C.5 and TS 3.3.A.2 for related BAST and RWST valves. The effect of continued operation with an inoperable BAST level channel is comparable to that associated with continued operation with an inoperable BAST supply valve and RWST supply valve.

Proposed Action 34, additionally, requires that if the inoperable BAST level channel cannot be restored to operable status within 72 hours, the unit must be brought to at least hot shutdown within the next 6 hours and to cold shutdown within the following 30 hours. These are standard shutdown times utilized throughout the licensee's current TSs.

d. Table TS.3.5-2B, Functional Unit 9b - BAST Automatic Actuation Logic and Actuation Relays

The licensee proposed a revision to Table TS.3.5-2B to add Functional Unit 9b, BAST Automatic Actuation Logic and Actuation Relays. Functional Unit 9b provides the requirements for the operability of the BAST to RWST transfer logic and actuation relay channels. The proposed TS required total number of operable channels is 2. The number of channels required to actuate the BAST to RWST transfer is 1. The proposed TS required operable channels for this functional unit are consistent in format and number with other actuation logic channels in current Table TS.3.5-2B. The proposed TS applicable operating modes, 1, 2, 3, and 4, are the same as those specified for the transfer interlocks in TS 3.2.B.5.

e. Table TS.3.5-2B, Action 35

Proposed Action 35 allows continued operation for 72 hours with an inoperable logic channel. The 72-hour allowance limits the time and associated risk incurred from operating with an inoperable logic channel while providing time for testing and repair or replacement of failed components. The 72-hour allowance for operation with an inoperable BAST to RWST logic channel is comparable to the allowed out-of-service times specified in TS 3.2.C.5 and TS 3.3.A.2.d for related BAST and RWST valves.

Proposed Action 35 also requires that if the inoperable logic channel cannot be restored to operable status within 72 hours, the unit must be brought to at least hot shutdown within the nest 6 hours and to cold shutdown within the following 30 hours. These are standard shutdown times utilized throughout the licensee's current TSs.

f. Table TS.3.5-2B, Action 36

Proposed Action 36 allows continued operation for 1 hour with both BAST and RWST transfer logic channels inoperable for surveillance testing. The 1-hour allowance limits the time and associated risk incurred from operation with both logic channels inoperable, while providing time for testing of the logic channels. The proposed 1-hour allowed outage time is comparable to the time allowed by the licensee's current TSs for the loss of function of other systems.

g. Bases 3.5, Instrumentation System

The proposed revision to TS Bases 3.5, "Instrumentation System" incorporates descriptions related to the proposed changes to TS 3.2.B and Table TS.3.5-2B and describes the automatic transfer of safety injection pump suction from the BAST to the RWST.

The staff finds that the above proposed TS changes provide appropriate limiting conditions for operation and action statements for operability of BAST lo-lo level instrumentation and for logic instrumentation for transfer of safety injection pump suction from the BAST to the RWST.

Based on the above evaluation, the staff concludes that the proposed changes to the Prairie Island TS to allow limited inoperability of the BAST level instrumentation and BAST to RWST transfer logic instrumentation are consistent with the current Prairie Island TSs for instrumentation performing similar safety functions and are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (63 FR 69345). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: B. Marcus

Date: March 17, 1999