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

April 16, 2001

Mr. Joel Sorensen
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: REMOVAL OF THE BORIC ACID STORAGE TANKS FROM THE SAFETY INJECTION SYSTEM (TAC NOS. MA8731 AND MA8732)

Dear Mr. Sorensen:

The Commission has issued the enclosed Amendment No. 156 to Facility Operating License No. DPR-42 and Amendment No. 147 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 17, 2000, as supplemented February 2, 2001.

The amendments change the TSs for removal of boric acid storage tanks from the safety injection (SI) system. These changes accomplish two objectives: (1) eliminate high concentration boric acid from the SI system and (2) align this specific TS section with the standard TSs.

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,

Tae Kim

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

Docket Nos. 50-282 and 50-306

Enclosures: 1. Amendment No. 156 to DPR-42
2. Amendment No. 147 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

NRR-058

April 16, 2001

Mr. Joel Sorensen
Site Vice President
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: REMOVAL OF THE BORIC ACID STORAGE TANKS FROM THE SAFETY INJECTION SYSTEM (TAC NOS. MA8731 AND MA8732)

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The amendments change the TSs for removal of boric acid storage tanks from the safety injection (SI) system. These changes accomplish two objectives: (1) eliminate high concentration boric acid from the SI system and (2) align this specific TS section with the standard TSs.

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Sincerely,
/RA/
Tae Kim, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures: 1. Amendment No. 156 to DPR-42
2. Amendment No. 147 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

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*Provided SE input by memo

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Prairie Island Nuclear Generating Plant,
Units 1 and 2

cc:

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 156
License No. DPR-42

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated April 17, 2000, as supplemented February 2, 2001, 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. 156 , 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 its date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 156

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
TS-ii	TS-ii
TS.3.2-1	--
TS.3.2-2	--
TS.3.3-1	TS.3.3-1
Table TS.3.5-2B (page 6 of 9)	Table TS.3.5-2B (page 6 of 9)
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Table TS.4.1-1C (page 1 of 4)	Table TS.4.1-1C (page 1 of 4)
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Table TS.4.1-2B (page 1 of 2)	Table TS.4.1-2B (page 1 of 2)
TS.4.5-3	TS.4.5-3
B.3.2-1	--
B.3.5-1	B.3.5-1
B.3.5-4	B.3.5-4

TABLE OF CONTENTS (Continued)

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.3.A.2 below):

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 2600 ppm.

   b. Each reactor coolant system accumulator shall be OPERABLE when reactor coolant system pressure is greater than 1000 psig.

      OPERABILITY requires:
      (1) The isolation valve is open
      (2) Volume is 1270 $\pm$20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of 740 $\pm$ 30 psig

   c. Two safety injection pumps are OPERABLE except as specified in Sections 3.3.A.3 and 3.3.A.4.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers 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. Deleted					

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: Deleted.

ACTION 35: Deleted.

ACTION 36: Deleted.

TABLE TS.4.1-1C (Page 1 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Control Rod Insertion Monitor	M	R	S/U[30]	N.A.	1, 2
2. Analog Rod Position	S	R	S/U[30]	N.A.	1, 2, 3[31], 4[31], 5[31]
3. Rod Position Deviation Monitor	M	N.A.	S/U[30]	N.A.	1, 2
4. Rod Position Bank Counters	S[32]	N.A.	N.A.	N.A.	1, 2, 3[31] 4[31], 5[31]
5. Deleted.					
6. Residual Heat Removal Pump Flow	S	R	N.A.	N.A.	4[37], 5[37], 6[37]
7. Deleted.					
8. Refueling Water Storage Tank Level	W	R	M	N.A.	1, 2, 3, 4
9. Deleted.					
10. Annulus Pressure (Vacuum Breaker)	N.A.	R	R	N.A.	See Note (39)
11. Auto Load Sequencers	N.A.	N.A.	M	N.A.	1, 2, 3, 4
12. Deleted.					

Table TS.4.1-1C
(Page 1 of 4)

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
W	Weekly
M	Monthly
Q	Quarterly
S/U	Prior to each reactor startup
Y	Yearly
R	Each Refueling Shutdown
N.A.	Not applicable.

TABLE NOTATION

(30) Prior to each startup following shutdown in excess of two days if not done in previous 30 days.

(31) When the reactor trip system breakers are closed and the control rod drive system is capable of rod withdrawal.

(32) Following rod motion in excess of six inches when the computer is out of service.

(33) Deleted.

(34) When either main steam isolation valve is open.

(35) Includes those instruments named in the emergency procedure.

(36) Except for containment hydrogen monitors and refueling water storage tank level which are separately specified in this table.

(37) When RHR is in operation.

(38) When the reactor coolant system average temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

(39) Whenever CONTAINMENT INTEGRITY is required.

TABLE TS.4.1-1C
(Page 4 of 4)

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	TEST	FREQUENCY
1.	RCS Gross Activity Determination	5/week
2.	RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3.	RCS Radiochemistry $\bar{E}$ determination	1/6 months(1) (when at power)
4.	RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5.	RCS Radiochemistry (2)	Monthly
6.	RCS Tritium Activity	Weekly
7.	Deleted	
8.	RCS Boron Concentration* (3)	2/Week (4)
9.	RWST Boron Concentration	Weekly
10.	Deleted	
11.	Caustic Standpipe NaOH Concentration	Monthly
12.	Accumulator Boron Concentration	Monthly
13.	Spent Fuel Pit Boron Concentration	Weekly

* Required at all times

B. Component Tests

1. Pumps

a. The safety injection pumps, residual heat removal pumps and containment spray pumps shall be tested pursuant to Specification 4.2. Acceptable levels of performance shall be that the pumps start and reach their required developed head on minimum recirculation flow and the control board indications and visual observations indicate that the pumps are operating properly for at least 15 minutes.

b. A test consisting of a manually-initiated start of each diesel engine, and assumption of load within one minute, shall be conducted monthly.

c. The vertical motor-driven cooling water pump shall be operated at quarterly intervals. An acceptable level of performance shall be that the pump starts and reaches its required developed head and the control board indications and visual observations indicate that the pump is operating properly for at least 15 minutes.

2. Containment Fan Motors

The Containment Fan Coil Units shall be run on low motor speed for at least 15 minutes at intervals of one month. Motor current shall be measured and compared to the nominal current expected for the test conditions.

3. Valves

a. Deleted.

b. The accumulator check valves will be checked for OPERABILITY during each refueling shutdown.

c. Deleted.

d. The spray chemical additive tank valves shall be tested in accordance with Section 4.2.

e. Actuation circuits for Cooling Water System valves that isolate non-essential equipment from the system shall be tested each refueling outage. Unit 1 SI actuation circuits for Train A and Train B valves shall be tested during Unit 1 refueling outages. Unit 2 SI actuation circuits for Train A and Train B valves shall be tested during Unit 2 refueling outages.

f. All motor-operated valves in the SIS, RHR, Containment Spray, Cooling Water, and Component Cooling Water System that are designed for operation during the safety injection or recirculation phase of emergency core cooling, shall be tested for OPERABILITY at each refueling shutdown.

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.

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 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

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.

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. DPR-60

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated April 17, 2000, as supplemented February 2, 2001, 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. 147 , 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 its date of its issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 147

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
TS-ii	TS-ii
TS.3.2-1	--
TS.3.2-2	--
TS.3.3-1	TS.3.3-1
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TS.4.5-3	TS.4.5-3
B.3.2-1	--
B.3.5-1	B.3.5-1
B.3.5-4	B.3.5-4

TABLE OF CONTENTS (Continued)

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

To define those limiting conditions that are necessary for operation of engineered safety features: (1) to remove decay heat from the core in an emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless the following conditions are satisfied (except as specified in 3.3.A.2 below):

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 2600 ppm.

   b. Each reactor coolant system accumulator shall be OPERABLE when reactor coolant system pressure is greater than 1000 psig.

      OPERABILITY requires:
      (1) The isolation valve is open
      (2) Volume is 1270 $\pm$20 cubic feet of borated water
      (3) A minimum boron concentration of 1900 ppm
      (4) A nitrogen cover pressure of 740 $\pm$ 30 psig

   c. Two safety injection pumps are OPERABLE except as specified in Sections 3.3.A.3 and 3.3.A.4.

   d. Two residual heat removal pumps are OPERABLE.

   e. Two residual heat exchangers 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. Deleted					

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: Deleted.

ACTION 35: Deleted.

ACTION 36: Deleted.

TABLE TS.4.1-1C (Page 1 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Control Rod Insertion Monitor	M	R	S/U(30)	N.A.	1, 2
2. Analog Rod Position	S	R	S/U(30)	N.A.	1, 2, 3(31), 4(31), 5(31)
3. Rod Position Deviation Monitor	M	N.A.	S/U(30)	N.A.	1, 2
4. Rod Position Bank Counters	S(32)	N.A.	N.A.	N.A.	1, 2, 3(31) 4(31), 5(31)
5. Deleted.					
6. Residual Heat Removal Pump Flow	S	R	N.A.	N.A.	4(37), 5(37), 6(37)
7. Deleted.					
8. Refueling Water Storage Tank Level	W	R	M	N.A.	1, 2, 3, 4
9. Deleted.					
10. Annulus Pressure (Vacuum Breaker)	N.A.	R	R	N.A.	See Note (39)
11. Auto Load Sequencers	N.A.	N.A.	M	N.A.	1, 2, 3, 4
12. Deleted.					

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
W	Weekly
M	Monthly
Q	Quarterly
S/U	Prior to each reactor startup
Y	Yearly
R	Each Refueling Shutdown
N.A.	Not applicable.

TABLE NOTATION

(30) Prior to each startup following shutdown in excess of two days if not done in previous 30 days.

(31) When the reactor trip system breakers are closed and the control rod drive system is capable of rod withdrawal.

(32) Following rod motion in excess of six inches when the computer is out of service.

(33) Deleted.

(34) When either main steam isolation valve is open.

(35) Includes those instruments named in the emergency procedure.

(36) Except for containment hydrogen monitors and refueling water storage tank level which are separately specified in this table.

(37) When RHR is in operation.

(38) When the reactor coolant system average temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

(39) Whenever CONTAINMENT INTEGRITY is required.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	TEST	FREQUENCY
1.	RCS Gross Activity Determination	5/week
2.	RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3.	RCS Radiochemistry $\overline{E}$ determination	1/6 months(1) (when at power)
4.	RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$ uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5.	RCS Radiochemistry (2)	Monthly
6.	RCS Tritium Activity	Weekly
7.	Deleted	
8.	RCS Boron Concentration* (3)	2/Week (4)
9.	RWST Boron Concentration	Weekly
10.	Deleted	
11.	Caustic Standpipe NaOH Concentration	Monthly
12.	Accumulator Boron Concentration	Monthly
13.	Spent Fuel Pit Boron Concentration	Weekly

* Required at all times

B. Component Tests

1. Pumps

a. The safety injection pumps, residual heat removal pumps and containment spray pumps shall be tested pursuant to Specification 4.2. Acceptable levels of performance shall be that the pumps start and reach their required developed head on minimum recirculation flow and the control board indications and visual observations indicate that the pumps are operating properly for at least 15 minutes.

b. A test consisting of a manually-initiated start of each diesel engine, and assumption of load within one minute, shall be conducted monthly.

c. The vertical motor-driven cooling water pump shall be operated at quarterly intervals. An acceptable level of performance shall be that the pump starts and reaches its required developed head and the control board indications and visual observations indicate that the pump is operating properly for at least 15 minutes.

2. Containment Fan Motors

The Containment Fan Coil Units shall be run on low motor speed for at least 15 minutes at intervals of one month. Motor current shall be measured and compared to the nominal current expected for the test conditions.

3. Valves

a. Deleted.

b. The accumulator check valves will be checked for OPERABILITY during each refueling shutdown.

c. Deleted.

d. The spray chemical additive tank valves shall be tested in accordance with Section 4.2.

e. Actuation circuits for Cooling Water System valves that isolate non-essential equipment from the system shall be tested each refueling outage. Unit 1 SI actuation circuits for Train A and Train B valves shall be tested during Unit 1 refueling outages. Unit 2 SI actuation circuits for Train A and Train B valves shall be tested during Unit 2 refueling outages.

f. All motor-operated valves in the SIS, RHR, Containment Spray, Cooling Water, and Component Cooling Water System that are designed for operation during the safety injection or recirculation phase of emergency core cooling, shall be tested for OPERABILITY at each refueling shutdown.

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.

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 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

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 156 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 147 TO FACILITY OPERATION LICENSE NO. DPR-60

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated April 17, 2000, Northern States Power Company (NSP) requested changes to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant, Units 1 and 2. NSP was subsequently succeeded by Nuclear Management Company, LLC (NMC or the licensee), as the licensed operator of Prairie Island, Units 1 and 2. By letter dated October 5, 2000, NMC requested the staff continue to process and disposition licensing actions previously docketed and requested by NSP. By letter dated February 2, 2000, NMC supplemented the April 17, 2000, application. The amendments would change the TSs for removal of boric acid storage tanks (BASTs) from the safety injection (SI) system. These changes accomplish two objectives: (1) eliminate high concentration boric acid from the SI system and (2) align this specific TS section with the standard TSs (STS). The February 2, 2001, supplement provided clarifying information that was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

Specifically, the proposed changes would modify TS 3.2, TS 3.3.A.1, TS 4.5.3, and TS Tables 3.5-2B, 4.1-1C, and 4.1-2B. These TS changes are related to removing the BASTs from the SI system; increasing the boron concentration requirement for the refueling water storage tank (RWST); removing the engineered safety feature actuation system instrumentation requirements for BAST level and the BAST automatic actuation logic and actuation relays; removing instrument surveillance requirements for charging flow, BAST level, volume control tank level, and boric acid makeup flow; removing the discussion of the BAST to RWST transfer logic; removing sampling requirements for BAST boron concentration; and removing requirements to test the SI suction valves from the BAST and the RWST. The TS Bases associated with these revised TSs are also modified accordingly.

2.0 EVALUATION

2.1 Supporting Analyses - The main steamline break (MSLB) accident is the only design-basis event that could be significantly affected by the proposed elimination of the BASTs as a safety-related high boron concentration fluid injection source for reactivity control. In the current SI system design, the BASTs with boron concentration of 20,000 ppm are normally aligned to the suction of the SI pumps. Following an MSLB accident and an SI actuation, when the BAST level falls to the lo-lo setpoint, an interlock automatically transfers the SI pumps' suction from the BASTs to the RWST with boron concentration of 2,500 ppm. The licensee has analyzed the MSLB accident without giving credit to the high boron concentration BASTs. The licensee's analysis uses the method documented in Topical Report NSPNAD-97002, "Northern States Power Company's Steam Line Break Methodology," Revision 1, which the NRC staff previously approved by letter dated January 21, 2000. The results of the licensee's safety analyses demonstrate that the high concentration boric acid in the BASTs is not required for the SI system to adequately mitigate a design-basis MSLB accident. The lower concentration boric acid of 2,500 ppm in the RWST is sufficient for mitigating the MSLB accident with acceptable results regarding core performance. The licensee-calculated minimum departure from nucleate boiling ratio (DNBR) for the most limiting MSLB accident is 1.46, which meets the 1.45 DNBR limit. Also, the licensee's analysis for a bounding large MSLB indicates that the maximum containment pressure is 44.89 psig, which is below its allowable limit of 46 psig, the peak containment temperature is 342 °F, which is below the previous analyzed value of 346.2 °F and the profile of the containment temperature is enveloped by the analysis currently contained within the environmental qualification (EQ) documentation. The licensee has evaluated the potential effects of its proposed changes on the loss-of-coolant accident (LOCA) accident. Since the licensee's current LOCA evaluation model does not credit the high boron concentration BASTs, the proposed removal of the BASTs from the SI system does not affect the LOCA calculation with respect to its acceptance criteria. However, the licensee proposed to change the minimum RWST boron concentration from the current 2,500 ppm to 2,600 ppm in order to increase the margin in long-term post-LOCA subcriticality. The NRC staff has reviewed the assumptions and the results of the licensee's analyses and has concluded that the assumptions used in these analyses are conservative and the results of these analyses meet the acceptance criteria for these events.

2.2 Proposed TS Changes to Support Removal of the BASTs

TS 3.2, Regarding the Chemical and Volume Control System - The licensee proposed to eliminate TS 3.2. Because the high concentration boric acid in the BASTs is unnecessary for accident mitigation, the requirements in current TS 3.2 do not meet any of the criteria in 10 CFR 50.36, which requires that TSs be established. In addition, this is consistent with the STS for Westinghouse plants in NUREG-1431. The NRC staff finds this change acceptable.

TS 3.3.A.1.a, Regarding Boron Concentration in RWST - The licensee proposed to increase the minimum boron concentration in the RWST from 2,500 ppm to 2,600 ppm. The licensee's analysis discussed in Section 2.1 above is based on an RWST boron concentration of 2,500 ppm for mitigation of design-basis accidents. However, the licensee indicates that it is desirable to slightly increase the RWST boron concentration for additional margin in maintaining subcriticality in long-term post-LOCA operation. The proposed change makes the TS more restrictive. The NRC staff finds this change acceptable.

TS Table 3.5-2B, Regarding ESF Actuation System Instrumentation - The licensee proposed to delete the TS requirements related to the BAST level instrumentation and automatic actuation logic and actuation relays associated with the BAST level. Since the licensee's safety analyses support the removal of the BASTs from the SI system, the NRC staff finds the proposed changes in TS Table 3.5-2B acceptable.

TS 4.1-1C, Regarding Miscellaneous Instrumentation Surveillance Requirements - The licensee proposed to delete the TS requirements related to BAST level, notation for transfer logic to the RWST, charging flow, volume control tank level, and boric acid make-up flow. The NRC staff has reviewed these proposed changes and concluded that they are consistent with the proposed removal of the BASTs from the SI system. In addition, these changes are consistent with the STS for Westinghouse plants in NUREG-1431. The NRC staff finds the proposed changes acceptable.

TS Table 4.1-2B, Regarding Sampling Tests of BAST Boron Concentrations - The licensee proposed to delete the TS requirements related to the sampling tests of BAST boron concentration. Since the licensee's analyses support the removal of the BASTs from the SI system, the requirements of sampling the BAST boron concentration are not necessary. The NRC staff finds the proposed change acceptable.

TS 4.5.3, Regarding Testing of SI System Suction Valves From BAST and RWST
The licensee proposed to delete test requirements for the SI suction valves from the BAST and the RWST. Since the BASTs are proposed to be removed from the SI system, the automatic function to transfer the water sources from the BASTs to the RWST is eliminated. The suction valve between the RWST and the SI pumps will be normally locked open and tagged in the proper position for injection per the requirements of TS 3.3.A.1.f. The NRC staff finds the proposed changes acceptable.

TS Bases

The licensee proposed changes to the TS Bases for the associated TS changes discussed above. The staff does not object to the changes proposed to the TS Bases.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves 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 amendment involves no significant hazards consideration and there has been no public comment on such finding (66 FR 13806). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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