Mr. Roger O. Andersor irector
Licensing and Managem... Issues
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

March 10,

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: LINE-ITEM TECHNICAL SPECIFICATION IMPROVEMENTS TO REDUCE SURVEILLANCE REQUIREMENTS FOR TESTING DURING POWER OPERATION AS RECOMMENDED BY GENERIC LETTER 93-05 (TAC NOS. M90459 AND M90460)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. DPR-42 and Amendment No. 109 to the Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated August 30, 1994.

The amendments revise Prairie Island TS Table 4.1-1C, "Miscellaneous Instrumentation Surveillance Requirements," TS Table 4.1-2A, "Minimum Frequencies for Equipment Tests," TS 4.3, "Primary Coolant System Pressure Isolation Valves," TS 4.4.I, "Electrical Hydrogen Recombiners," TS 4.5.A.2.b, "Containment Spray System," TSs 4.8.A.1, 4.8.A.2, and Footnote, "Auxiliary Feedwater System," and Bases 4.8, "Steam and Power Conversion Systems." These changes implement recommended changes from Generic Letter (GL) 93-05, "Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation." Specifically, the amendments implement TS changes corresponding to the following GL 93-05 line numbers: 7.4, 4.2, 6.1, 8.5, 8.1, and 9.1.

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

Charles Thomas, Acting Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 116 to DPR-42
2. Amendment No. 109 to DPR-60
3. Safety Evaluation

cc w/encls: See next page

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DATED: March 10, 1995

AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
J. Hannon
C. Jamerson
C. Thomas (2)
OGC-WF
H. Abelson
G. Hill, IRM (4)
C. Grimes, O-11F23
ACRS (4)
OPA
OC/LFDCB
W. Kropp, RIII
SEDB

cc: Plant Service list

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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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
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 August 30, 1994, 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. 116 , 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

Kevin A. Conway for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 10, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 116

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 TABLE 4.1-1C (Page 2 of 4)	TS TABLE 4.1-1C (Page 2 of 4)
TS TABLE 4.1-2A	TS TABLE 4.1-2A
TS 4.3-1	TS 4.3-1
TS 4.4-5	TS 4.4-5
TS 4.5-1	TS 4.5-1
TS 4.8-1	TS 4.8-1
TS BASES 4.8-1	TS BASES 4.8-1

TABLE TS.4.1-1C (Page 2 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Containment Sump A, B and C Level	N.A.	R	R	N.A.	1, 2, 3, 4
14. Deleted					
15. Turbine First Stage Pressure	S	R	Q	N.A.	1
16. Emergency Plan Radiation Instruments [35]	M	R	M	N.A.	1, 2, 3, 4, 5, 6
17. Seismic Monitors	R	R	N.A.	N.A.	1, 2, 3, 4, 5, 6
18. Coolant Flow - RTD Bypass Flowmeter	S	R	M	N.A.	1, 2, 3[34]
19. CRDM Cooling Shroud Exhaust Air Temperature	S	N.A.	R	N.A.	1, 2, 3[31], 4[31], 5[31]
20. Reactor Gap Exhaust Air Temperature	S	N.A.	R	N.A.	1, 2, 3, 4
21. Post-Accident Monitoring Instruments (Table TS.3.15-1)[36]	M	R	N.A.	N.A.	1, 2
22. Post-Accident Monitoring Radiation Instruments (Table TS.3.15-2)	D	R	M	N.A.	1, 2

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	FSAR Sect. Reference
1. Control Rod Assemblies	Rod Drop Times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods	7
2. Control Rod Assemblies	Partial movement of all rods	Every Quarter	7
3. Pressurizer Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
5. Reactor Cavity	Water Level	Prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.	
6. Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1).(b).2 or 3.1.A.2.c.(1).(b).3.	-
7. Pressurizer PORVs	Functional	Every 18 months	-
8. Deleted			
9. Primary System Leakage	Evaluate	Daily	4
10. Deleted			
11. Turbine stop valves, governor valves, and intercept valves. (Part of turbine overspeed protection)	Functional	See (1)	10

(1) Turbine stop valves, governor valves and intercept valves are to be tested at a frequency consistent with the methodology presented in WCAP-11525 "Probabilistic Evaluation of Reduction in Turbine Valve test Frequency", and in accordance with the established NRC acceptance criteria for the probability of a turbine missile ejection incident of 1.0×10^{-5} per year. In no case shall the turbine valve test interval exceed one year.

4.3 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Applicability

Applies to the surveillance performed on the primary coolant system pressure isolation valves to verify operability.

Objective

To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

Specification

Periodic leakage testing of each of the following valves shall be individually accomplished prior to resuming power operation after each time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 7 days or more if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed:

System	Valve Number Unit No. 1	Valve Number Unit No. 2	Maximum Allowable Leakage (*)(**)
Low-Pressure SI to Upper Plenum	SI-9-6	2SI-9-6	≤ 5 gpm
	SI-9-4	2SI-9-4	≤ 5 gpm
	SI-9-5	2SI-9-5	≤ 5 gpm
	SI-9-3	2SI-9-3	≤ 5 gpm
RHR to Loop B Accumulator Inj Line	SI-6-2	2SI-6-2	≤ 5 gpm

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

NOTES:

* 1. Leakage rates less than or equal to one gpm are acceptable.

2. Leakage rates greater than one, but less than or equal to five gpm are considered acceptable if the latest measured rate has not exceeded the previous measured rate by an amount which reduces the margin to five gpm by 50% or more. Otherwise the leakage rate is considered unacceptable.

3. Leakage rates greater than five gpm are considered unacceptable.

** Minimum differential test pressure shall not be less than 150 psid.

E. Containment Isolation Valves

During each refueling shutdown, the containment isolation valves, shield building ventilation valves, and the auxiliary building normal ventilation system isolation valves shall be tested for operability by applying a simulated accident signal to them.

F. Post Accident Containment Ventilation System

During each refueling shutdown, the operability of system recirculating fans and valves, including actuation and indication, shall be demonstrated.

G. Containment and Shield Building Air Temperature

Prior to establishing reactor conditions requiring containment integrity, the average air temperature difference between the containment and its associated Shield Building shall be verified to be within acceptable limits.

H. Containment Shell Temperature

Prior to establishing reactor conditions requiring containment integrity, the temperature of the containment vessel wall shall be verified to be within acceptable limits.

I. Electric Hydrogen Recombiners

Each hydrogen recombiner train shall be demonstrated Operable at least once each refueling interval by:

a. Verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60kw.

b. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

c. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

4.5 ENGINEERED SAFETY FEATURES

Applicability

Applies to testing of the Emergency Core Cooling System and the Containment Cooling Systems.

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required

Specification

A. System Tests

1. Safety Injection System

   a. System tests shall be performed during each reactor refueling shutdown. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The safety injection and residual heat removal pumps need not be operable for this test.

   b. The test will be considered satisfactory if control board indications and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, the appropriate pump breakers have opened and closed, and all automatic valves have been placed in the proper position required to establish a safety injection flow path to the reactor coolant system.

2. Containment Spray System

   a. System tests shall be performed during each reactor refueling shutdown. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

   b. The spray nozzles shall be checked for proper functioning at least every ten years.

   c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater, Steam Generator Power Operated Relief Valves, and Steam Exclusion Systems.

Objective

To verify the OPERABILITY of the steam and power conversion systems required for emergency shutdown cooling of the plant.

Specification

A. Auxiliary Feedwater System

1. Each auxiliary feedwater pump* shall be started semi-quarterly on a STAGGERED TEST BASIS and full flow to the steam generators shall be demonstrated once every refueling shutdown.

2. Deleted.

3. The auxiliary feedwater pumps discharge valves shall be tested by operator action in accordance with Section 4.2.

4. Motor-operated valves required to function during accident conditions shall be tested in accordance with Section 4.2.

5. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

6. During POWER OPERATION, for the manual valves outside containment, that could reduce auxiliary feedwater flow, if improperly positioned, to less than assumed in the accident analysis, monthly inspections are required to verify the valves are locked in the proper position required for emergency use.

7. After each COLD SHUTDOWN and prior to exceeding 10% power, a test is required to verify the normal flow path from the primary auxiliary feedwater source to the steam generators. This test may consist of maintaining steam generator level during startup with the auxiliary feed pumps.

8. At least once every 18 months during shutdown verify that each pump starts as designed automatically and each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

*If the test for a steam turbine-driven pump comes due during a reactor shutdown the test shall be performed within 24 hours of entering POWER OPERATION.

4.8 STEAM AND POWER CONVERSION SYSTEMS

Bases

The Surveillance Frequency for the auxiliary feedwater pumps specifies that the pumps shall be started semi-quarterly on a STAGGERED TEST BASIS. Per the definition of STAGGERED TEST BASIS, the Surveillance Frequency interval is semi-quarterly and the number of trains (channels) is 2 (n=2). Therefore, STAGGERED TEST BASIS requires one pump shall be tested each semi-quarter such that after two Surveillance Frequency intervals, i.e., one quarter, both trains will have been tested.

Quarterly testing of each auxiliary feedwater pump, valve inspections in accordance with Section 4.2, and startup flow verification provide assurance that the Auxiliary Feedwater System will meet emergency demand requirements. The full flow test is done once a cycle associated with the refueling shutdown to minimize the thermal shock to the auxiliary feedwater piping. The discharge valves of the pumps are normally open, as are the suction valves from the condensate storage tanks. Proper opening of the steam admission valve on each turbine-driven pump will be demonstrated each time a turbine-driven pump is tested. Ventilation system isolation dampers required to function for the postulated rupture of a high energy line will also be tested.

At 18-month intervals, pump starting and valve positioning is verified using test signals to simulate each of the automatic actuation parameters.

Reference

USAR, Sections 11.9, 14, and 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 NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 109
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 August 30, 1994, 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. 109, 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

Kevin A. Connaughton for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 10, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 109

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 TABLE 4.1-1C (Page 2 of 4)	TS TABLE 4.1-1C (Page 2 of 4)
TS TABLE 4.1-2A	TS TABLE 4.1-2A
TS 4.3-1	TS 4.3-1
TS 4.4-5	TS 4.4-5
TS 4.5-1	TS 4.5-1
TS 4.8-1	TS 4.8-1
TS BASES 4.8-1	TS BASES 4.8-1

TABLE TS.4.1-1C (Page 2 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Containment Sump A, B and C Level	N.A.	R	R	N.A.	1, 2, 3, 4
14. Deleted					
15. Turbine First Stage Pressure	S	R	Q	N.A.	1
16. Emergency Plan Radiation Instruments (35)	M	R	M	N.A.	1, 2, 3, 4, 5, 6
17. Seismic Monitors	R	R	N.A.	N.A.	1, 2, 3, 4, 5, 6
18. Coolant Flow - RTD Bypass Flowmeter	S	R	M	N.A.	1, 2, 3(34)
19. CRDM Cooling Shroud Exhaust Air Temperature	S	N.A.	R	N.A.	1, 2, 3(31), 4(31), 5(31)
20. Reactor Gap Exhaust Air Temperature	S	N.A.	R	N.A.	1, 2, 3, 4
21. Post-Accident Monitoring Instruments (Table TS.3.15-1)(36)	M	R	N.A.	N.A.	1, 2
22. Post-Accident Monitoring Radiation Instruments (Table TS.3.15-2)	D	R	M	N.A.	1, 2

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Test	Frequency	FSAR Sect. Reference
1. Control Rod Assemblies	Rod Drop Times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods	7
2. Control Rod Assemblies	Partial movement of all rods	Every Quarter	7
3. Pressurizer Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
4. Main Steam Safety Valves	Set point	Per ASME Code, Section XI Inservice Testing Program	-
5. Reactor Cavity	Water Level	Prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.	
6. Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1).(b).2 or 3.1.A.2.c.(1).(b).3.	-
7. Pressurizer PORVs	Functional	Every 18 months	-
8. Deleted			
9. Primary System Leakage	Evaluate	Daily	4
10. Deleted			
11. Turbine stop valves, governor valves, and intercept valves. (Part of turbine overspeed protection)	Functional	See (1)	10

(1) Turbine stop valves, governor valves and intercept valves are to be tested at a frequency consistent with the methodology presented in WCAP-11525 "Probabilistic Evaluation of Reduction in Turbine Valve test Frequency", and in accordance with the established NRC acceptance criteria for the probability of a turbine missile ejection incident of 1.0×10^{-5} per year. In no case shall the turbine valve test interval exceed one year.

4.3 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Applicability

Applies to the surveillance performed on the primary coolant system pressure isolation valves to verify operability.

Objective

To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

Specification

Periodic leakage testing of each of the following valves shall be individually accomplished prior to resuming power operation after each time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 7 days or more if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed:

System	Valve Number Unit No. 1	Valve Number Unit No. 2	Maximum Allowable Leakage (*)(**)
Low-Pressure SI to Upper Plenum	SI-9-6	2SI-9-6	≤ 5 gpm
	SI-9-4	2SI-9-4	≤ 5 gpm
	SI-9-5	2SI-9-5	≤ 5 gpm
	SI-9-3	2SI-9-3	≤ 5 gpm
RHR to Loop B Accumulator Inj Line	SI-6-2	2SI-6-2	≤ 5 gpm

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

NOTES:

* 1. Leakage rates less than or equal to one gpm are acceptable.

2. Leakage rates greater than one, but less than or equal to five gpm are considered acceptable if the latest measured rate has not exceeded the previous measured rate by an amount which reduces the margin to five gpm by 50% or more. Otherwise the leakage rate is considered unacceptable.

3. Leakage rates greater than five gpm are considered unacceptable.

** Minimum differential test pressure shall not be less than 150 psid.

E. Containment Isolation Valves

During each refueling shutdown, the containment isolation valves, shield building ventilation valves, and the auxiliary building normal ventilation system isolation valves shall be tested for operability by applying a simulated accident signal to them.

F. Post Accident Containment Ventilation System

During each refueling shutdown, the operability of system recirculating fans and valves, including actuation and indication, shall be demonstrated.

G. Containment and Shield Building Air Temperature

Prior to establishing reactor conditions requiring containment integrity, the average air temperature difference between the containment and its associated Shield Building shall be verified to be within acceptable limits.

H. Containment Shell Temperature

Prior to establishing reactor conditions requiring containment integrity, the temperature of the containment vessel wall shall be verified to be within acceptable limits.

I. Electric Hydrogen Recombiners

Each hydrogen recombiner train shall be demonstrated Operable at least once each refueling interval by:

a. Verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60kw.

b. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

c. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

4.5 ENGINEERED SAFETY FEATURES

Applicability

Applies to testing of the Emergency Core Cooling System and the Containment Cooling Systems.

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required

Specification

A. System Tests

1. Safety Injection System

    a. System tests shall be performed during each reactor refueling shutdown. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The safety injection and residual heat removal pumps need not be operable for this test.

    b. The test will be considered satisfactory if control board indications and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, the appropriate pump breakers have opened and closed, and all automatic valves have been placed in the proper position required to establish a safety injection flow path to the reactor coolant system.

2. Containment Spray System

    a. System tests shall be performed during each reactor refueling shutdown. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

    b. The spray nozzles shall be checked for proper functioning at least every ten years.

    c. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

4.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater, Steam Generator Power Operated Relief Valves, and Steam Exclusion Systems.

Objective

To verify the OPERABILITY of the steam and power conversion systems required for emergency shutdown cooling of the plant.

Specification

A. Auxiliary Feedwater System

1. Each auxiliary feedwater pump* shall be started semi-quarterly on a STAGGERED TEST BASIS and full flow to the steam generators shall be demonstrated once every refueling shutdown.

2. Deleted.

3. The auxiliary feedwater pumps discharge valves shall be tested by operator action in accordance with Section 4.2.

4. Motor-operated valves required to function during accident conditions shall be tested in accordance with Section 4.2.

5. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

6. During POWER OPERATION, for the manual valves outside containment, that could reduce auxiliary feedwater flow, if improperly positioned, to less than assumed in the accident analysis, monthly inspections are required to verify the valves are locked in the proper position required for emergency use.

7. After each COLD SHUTDOWN and prior to exceeding 10% power, a test is required to verify the normal flow path from the primary auxiliary feedwater source to the steam generators. This test may consist of maintaining steam generator level during startup with the auxiliary feed pumps.

8. At least once every 18 months during shutdown verify that each pump starts as designed automatically and each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

*If the test for a steam turbine-driven pump comes due during a reactor shutdown the test shall be performed within 24 hours of entering POWER OPERATION.

4.8 STEAM AND POWER CONVERSION SYSTEMS

Bases

The Surveillance Frequency for the auxiliary feedwater pumps specifies that the pumps shall be started semi-quarterly on a STAGGERED TEST BASIS. Per the definition of STAGGERED TEST BASIS, the Surveillance Frequency interval is semi-quarterly and the number of trains (channels) is 2 (n=2). Therefore, STAGGERED TEST BASIS requires one pump shall be tested each semi-quarter such that after two Surveillance Frequency intervals, i.e., one quarter, both trains will have been tested.

Quarterly testing of each auxiliary feedwater pump, valve inspections in accordance with Section 4.2, and startup flow verification provide assurance that the Auxiliary Feedwater System will meet emergency demand requirements. The full flow test is done once a cycle associated with the refueling shutdown to minimize the thermal shock to the auxiliary feedwater piping. The discharge valves of the pumps are normally open, as are the suction valves from the condensate storage tanks. Proper opening of the steam admission valve on each turbine-driven pump will be demonstrated each time a turbine-driven pump is tested. Ventilation system isolation dampers required to function for the postulated rupture of a high energy line will also be tested.

At 18-month intervals, pump starting and valve positioning is verified using test signals to simulate each of the automatic actuation parameters.

Reference

USAR, Sections 11.9, 14, and Appendix I.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 116 AND 109 TO

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated August 30, 1994, the Northern States Power Company (the licensee) submitted a request for change to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. Specifically, the request would revise TS Section Table 4.1-1C, "Miscellaneous Instrumentation Surveillance Requirements," TS Table 4.1-2A, "Minimum Frequencies for Equipment Tests," TS 4.3, "Primary Coolant System Pressure Isolation Valves," and TSs 4.4.I, 4.4.I.a, 4.4.I.b, 4.4.I.b.1, 4.4.I.b.2, and 4.4.I.b.3, "Electrical Hydrogen Recombiners," TS 4.5.A.2.b, "Containment Spray System," TSs 4.8.A.1, 4.8.A.2, and Footnote, "Auxiliary Feedwater System," and Bases 4.8, "Steam and Power Conversion Systems." These revisions would implement recommended changes from Generic Letter (GL) 93-05, "Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation." Specifically, the amendments would implement TS changes corresponding to the following GL 93-05 line numbers: 7.4, 4.2, 6.1, 8.5, 8.1, and 9.1.

2.0 BACKGROUND

NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," December 1992, reported the TS line-item improvements that were identified by the NRC staff. The TS improvements were based on an NRC study of surveillance requirements (SRs) and included information provided by licensee personnel that plan, manage, and perform surveillances. The study included insights from a qualitative risk assessment of SRs bases on the standard TS for Westinghouse plants and the TS for the Edwin I. Hatch Nuclear Plant, Unit 2. The staff examined operational data from licensee events reports, the nuclear plant reliability data system (NPRDS), and other sources to assess the effect of TS SRs on plant operation. The staff evaluated the effect of longer surveillance intervals to reduce the possibility for plant transients, wear on equipment, personnel radiation exposure, and burden on personnel resources. Finally, the staff considered surveillance activities for which the safety benefits are small and not justified when compared to the effects of these activities on the safety of personnel and the plant. The NRC staff issued guidance on the proposed TS changes to all holders of operating licenses or construction permits for nuclear power reactors in GL 93-05, September 27, 1993.

3.0 EVALUATION

The staff has evaluated the licensee's proposed TS SR modifications as described below:

Table 4.1-1C, "Miscellaneous Instrumentation Surveillance Requirements." Delete Item 14, "Accumulator Level and Pressure" and corresponding frequency interval designations and place them into the licensee-controlled test procedures.

This TS modification implements GL 93-05, Item 7.4, Accumulator Water Level and Pressure Channel Surveillance Requirements (PWR) and is consistent with industry recognition that accumulator instrumentation operability is not directly related to the capability of the accumulators to perform their safety function.

Table 4.1-2A, "Minimum Frequencies For Equipment Tests." Revise the frequency for partial movement of all control rod assemblies from every 2 weeks to once per quarter.

This TS modification implements GL 93-05, Item 4.2, Control Rod Movement Test.

TS SR 4.3, "Primary Coolant System Pressure Isolation Valves." Under Specification heading, extend the amount of time the plant can be shut down before pressure isolation valve (PIV) testing will be required from 72 hours to 7 days.

This TS modification implements GL 93-05, Item 6.1, Reactor Coolant System Isolation Valves (PWR).

TS SRs 4.4.I, 4.4.I.a, 4.4.I.b, 4.4.I.b.1, 4.4.I.b.2, and 4.4.I.b.3, "Electrical Hydrogen Recombiners." Revise the containment hydrogen recombiner testing surveillance frequency from every 6 months to every refueling interval. Delete the specific requirement to perform CHANNEL CALIBRATION of recombiner instruments and control circuits and relocate to licensee-controlled procedures. Delete the requirement to sequentially perform the resistance to ground test following the function test. This latter requirement has no technical basis and is unnecessarily restrictive. Its removal would bring the TS into conformance with the Standard TS.

This TS modification implements GL 93-05, Item 8.5, Hydrogen Recombiner (PWR, BWR).

TS SR 4.5.A.2.b, "Containment Spray System." Revise the containment spray system nozzle testing surveillance frequency from once every 5 years to once every 10 years.

This TS modification implements GL 93-05, Item 8.1, Containment Spray System (PWR).

TS SR 4.8.A.1, 4.8.A.2, and Footnote, "Auxiliary Feedwater System." Revise the testing frequency for the auxiliary feedwater pumps from intervals of 1 month to semi-quarterly on a STAGGERED TEST BASIS.

This TS modification implements GL 93-05, Item 9.1, Auxiliary Feedwater Pump and System Testing (PWR).

TS Bases 4.8, "Steam and Power Conversion Systems." Revise the Bases to include testing frequency for the auxiliary feedwater pumps from intervals of 1 month to semi-quarterly on a STAGGERED TEST BASIS. This is an administrative change to bring the Bases into conformance with the TS.

The proposed TS modifications are consistent with the guidance provided in GL 93-05. This guidance is based on the NRC staff findings and recommendations stated in NUREG-1366. NUREG-1366 recognized that testing is important to periodically verify that systems, structures, and components are available to perform their safety functions. Testing is especially critical to reveal degradation and failures that occur while equipment is in standby mode. The study did find that, while most testing at power is important, safety can be improved, equipment degradation decreased, and an unnecessary burden on personnel resources eliminated by reducing the amount of testing that TS required during power operation. However, only a small fraction of the TS surveillance intervals warranted relaxation. In addition, the licensee stated that the proposed TS changes are compatible with plant operating experience. The staff concludes that the proposed TS changes do not adversely affect plant safety and will result in a net benefit to the safe operation of the facility, and, therefore, are acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments 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 (60 FR 6305). 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.

6.0 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.

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