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

July 9, 1992

Docket Nos. 50-282
and 50-306

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - AMENDMENT NOS. 99 AND 92 TO FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60 (TAC NOS. M81883 AND M81884)

The Commission has issued the enclosed Amendment No. 99to Facility Operating License No. DPR-42 and Amendment No. 92 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 in response to your application dated October 4, 1991, as supplemented by your letter of December 16, 1991.

The amendments revise Technical Specification Section 3.8.B and its associated Bases to remove the restriction related to cask handling; add a new Section 4.19 and associated Bases which establish surveillance requirements for the Auxiliary Building crane lifting devices; and revise Section 5.6 to remove references to the spent fuel cask drop analysis and mitigation design features, and incorporate a new paragraph which states that spent fuel casks will be handled by a single-failure-proof handling system.

The amendments also make several changes of an administrative nature in Technical Specification Sections 3.8.B, 5.6, and in Table TS 4.1-2B in order to accommodate placement of spent fuel storage casks in the spent fuel pool, and to discuss the Bases for spent fuel boron requirements to maintain the boron concentration level, provide an action statement if boron concentration falls below required levels, and require a weekly verification of the boron concentration.

DF01

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,

/s/

Armando Masciantonio, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 99 to DPR-42
2. Amendment No. 92 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

W/ changes as noted.

APH
6/25/92

OFC	:LA:PDIII-1	PM:PDIII-1 asm	SPLB	D:PDIII-1	OGC	SRXB
NAME	:MShuttleworth	AMasciantonio:sw:	CMcCracken	LBMarsh		RJones
DATE	:[illegible]/[illegible]/92	5/19/92	5/14/92	[illegible]/[illegible]/92	5/21/92	5/18/92

OFFICIAL RECORD COPY
Document Name:C:81883-4.SHO

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,

A.S. [signature]

Armando Masciantonio, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 99 to DPR-42
2. Amendment No. 92 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

cc:

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

Mr. E. L. Watzl, Site General Manger
Prairie Island Nuclear Generating
Plant
Northern States Power Company
Route 2
Welch, Minnesota 55089

Lisa R. Tiegel
Assistant Attorney General
Environmental Protection Division
Suite 200
520 Lafayette Road
St. Paul, Minnesota 55155

U.S. Nuclear Regulatory Commission
Resident Inspector Office
1719 Wakonade Drive
East Welch, Minnesota 55089

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. William Miller, Auditor
Goodhue County Courthouse
Red Wing, Minnesota 55066

Kris Sanda, Commissioner
Director of Public Service
790 American Center Building
150 East Kellogg Boulevard
St. Paul, Minnesota 55101

Amy C. Kvalseth
Special Assistant Attorney General
1100 Bremer Tower
7th Place and Minnesota Street
St. Paul, Minnesota 55101

DATED: July 9, 1992

AMENDMENT NO.99 TO FACILITY OPERATING LICENSE NO. DPR-42-PRAIRIE ISLAND UNIT 1
AMENDMENT NO.92 TO FACILITY OPERATING LICENSE NO. DPR-60-PRAIRIE ISLAND UNIT 2

Docket File
NRC & Local PDRs
PDIII-1 Reading
PI Plant File
B. Boger, 13/E/4
J. Zwolinski, 13/H/24
L. Marsh
M. Shuttleworth
A. Masciantonio
OGC-WF
D. Hagan, 3206 MNBB
G. Hill (8), P-137
Wanda Jones, MNBB-3701
C. Grimes, 11/F/23
ACRS (10)
OC/LFMB
W. Shafer, R-III
N. Wagner
L. Kopp
J. Schneider, NMSS
cc: Plant Service list

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
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 October 4, 1991, 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.99, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 9, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 99

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 marginal lines indicating the area of change.

REMOVE	INSERT
TS-iii	TS-iii
TS-vi	TS-vi
TS-vii	TS-vii
TS-xi	TS-xi
TS.3.8-3	TS.3.8-3
Table TS.4.1-2B (page 1 of 2)	Table TS.4.1-2B (page 1 of 2)
Table TS.4.1-2B (page 2 of 2)	Table TS.4.1-2B (page 2 of 2)
----	TS.4.19-1
TS.5.6-1	TS.5.6-1
TS.5.6-2	TS.5.6-2
TS.5.6-3	----
B.3.8-1	B.3.8-1
B.3.8-2	B.3.8-2
B.3.8-3	----
----	B.4.19-1

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (continued)

3.8.B. <u>Fuel Handling Operations</u>

1. During fuel handling operations the following conditions shall be satisfied:

    a. Radiation levels in the spent fuel storage pool area shall be monitored continuously during fuel handling operations.

    b. Prior to fuel handling operations, fuel-handling cranes shall be load-tested for OPERABILITY of limit switches, interlocks and alarms.

    c. A minimum boron concentration of 1800 ppm shall be maintained in the spent fuel pool whenever a spent fuel cask containing fuel is located in the spent fuel pool.

2. If any of the conditions in 3.8.B.1, above, cannot be met, suspend all fuel handling operations and initiate the actions necessary to re-establish compliance with the requirements of 3.8.B.1.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
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. RCS Chemistry (Cl*,F*, O_2)	5/Week	
8. RCS Boron Concentration*(3)	2/Week (4)	9.2
9. RWST Boron Concentration	Weekly	
10. Boric Acid Tanks Boron Concentration	2/Week	
11. Caustic Standpipe NaOH Concentration	Monthly	6.4
12. Accumulator Boron Concentration	Monthly	6
13. Spent Fuel Pit Boron Concentration	Monthly (7)	9.5.5

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST		FREQUENCY	FSAR Section Reference
14.	Secondary Coolant Gross Beta-Gamma activity	Weekly	
15.	Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)	
16.	Secondary Coolant Chemistry		
	pH	5/week (6)	
	pH Control Additive	5/week (6)	
	Sodium	5/week (6)	

Notes:

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.
2. To determine activity of corrosion products having a half-life greater than 30 minutes.
3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.
4. The maximum interval between analyses shall not exceed 5 days.
5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.
6. The maximum interval between analyses shall not exceed 3 days.
7. The minimum spent fuel pool boron concentration from Specification 3.8.B.1.b shall be verified by chemical analysis weekly while a spent fuel cask containing fuel is located in the spent fuel pool.

* See Specification 4.1.D

4.19 Auxiliary Building Crane Lifting Devices

Applicability

Applies to surveillance requirements for the auxiliary building crane special lifting devices and slings before handling heavy loads carried over safe shutdown equipment or spent fuel in the spent fuel pool.

Objective

To verify that special lifting devices and slings used in conjunction with the auxiliary building crane are operable prior to their use in supporting heavy loads over safe shutdown equipment or spent fuel in the spent fuel pool.

Specification

Slings and special lifting devices which will be used in supporting heavy loads from the auxiliary building crane shall be visually inspected and verified OPERABLE within 7 days prior to their use in handling heavy loads over safe shutdown equipment or spent fuel in the spent fuel pool.

5.6 FUEL HANDLING

A. Criticality Consideration

The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I (seismic) structures. The spent fuel pit has a stainless steel liner to ensure against loss of water (Reference 1).

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The design of the new fuel storage pit and racks (Reference 1) ensures a new fuel pit K_{eff} of less than or equal to 0.95, including uncertainties, even if unborated water were used to fill the pit. The new fuel rack configuration also ensures K_{eff} less than or equal to 0.98, including uncertainties, even if the new fuel racks were accidentally filled with a low density moderator which resulted in optimum low density moderation conditions. Fuel stored in the new fuel storage racks will have a maximum enrichment of 4.25 weight percent U-235.

The spent fuel storage rack design (Reference 1) and the limitations on the storage of low burnup fuel contained in Technical Specification Section 3.8.E ensure a spent fuel pool K_{eff} of less than or equal to 0.95, including uncertainties. The maximum enrichment of fuel to be stored in the spent fuel pool will be 4.25 weight percent U-235.

Fuel will not be inserted into a spent fuel cask in the pool, unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask. The criticality analysis for the TN-40 spent fuel storage cask was based on fresh fuel enriched to 3.85 weight percent U-235.

B. Spent Fuel Storage Structure

The spent fuel storage pool is enclosed with a reinforced concrete building having 12- to 18-inch thick walls and roof (Reference 1). The pool and pool enclosure are Class I (seismic) structures that afford protection against loss of integrity from postulated tornado missiles. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The bottoms of the slots are above the tops of the active fuel in the fuel assemblies which will be stored vertically in specially constructed racks.

The spent fuel pool has a reinforced concrete bottom slab nearly 6 feet thick and has been designed to minimize loss of water due to a dropped cask accident. Piping to the pool is arranged so that failure of any pipe cannot drain the pool below the tops of the stored fuel assemblies.

C. Fuel Handling

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after post-irradiation cooling. The system consists of the refueling cavity, the fuel transfer system, the spent fuel storage pit, and the spent fuel cask transfer system.

Major components of the fuel handling system are the manipulation crane, the spent fuel pool bridge, the auxiliary building crane, the fuel transfer system, the spent fuel storage racks, the spent fuel cask, and the rod cluster control changing fixture. The reactor vessel stud tensioner, the reactor vessel head lifting device, and the reactor internals lifting device are used for preparing the reactor for refueling and for assembling the reactor after refueling.

Upon arrival in the storage pit, spent fuel will be removed from the transfer system and placed, one assembly at a time, in storage racks using a long-handled manual tool suspended from the spent fuel pit bridge crane. After sufficient decay, the fuel will be loaded into storage casks for storage in the Independent Spent Fuel Storage Installation or into shipping casks for removal from the site. The casks will be handled by the auxiliary building crane.

Spent fuel casks will be handled by a single failure proof handling system meeting the requirements of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980. The auxiliary building crane has been upgraded to conform with the single failure proof requirements of Section 5.1.6 of NUREG-0612. The auxiliary building crane is designed to not allow a load drop as a result of any single failure. The improved reliability of the auxiliary building crane is achieved through increased factors of safety and through redundancy or duality in certain active components.

D. Spent Fuel Storage Capacity

The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, provides up to 1582 storage locations. The southeast corner of the small pool (pool no. 1) also serves as the cask lay down area. During times when the cask is being used, four racks are removed from the small pool. With the four storage racks in the southeast corner of pool 1 removed, a total of 1386 storage locations are provided. To allow insertion of a spent fuel cask, total storage is limited to 1386 assemblies, not including those assemblies which can be returned to the reactor.

Reference

1. USAR, Section 10.2

3.8 REFUELING AND FUEL HANDLING

Bases

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the precautions specified above, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during CORE ALTERATIONS that would result in a hazard to public health and safety (Reference 1). Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

Under rodded and unrodded conditions, the K_{eff} of the reactor must be less than or equal to 0.95 and the boron concentration must be greater than or equal to 2000 ppm. Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. 3.8.A.1.h allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

No movement of fuel in the reactor is permitted until the reactor has been subcritical for at least 100 hours to permit decay of the fission products in the fuel. The delay time is consistent with the fuel handling accident analysis (Reference 2).

Fuel will not be inserted into a spent fuel cask unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask.

The number of recently discharged assemblies in Pool No. 1 has been limited to 45 to provide assurance that in the event of loss of pool cooling capability, at least eight hours are available under worst case conditions to make repairs until the onset of boiling.

The Spent Fuel Pool Special Ventilation System (Reference 3) is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation System is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

3.8 REFUELING AND FUEL HANDLING

Bases continued

During movement of irradiated fuel assemblies or control rods, a water level of 23 feet is maintained to provide sufficient shielding.

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The water level may also be lowered below 20 feet for upper internals removal/replacement. The basis for these allowance(s) are (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching/unlatching and upper internals removal/replacement the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

The requirements for the storage of low burnup fuel in the spent fuel pool ensure that the spent fuel pool will remain subcritical during fuel storage. Fuel stored in the spent fuel pool will be limited to a maximum enrichment of 4.25 weight percent U-235. It has been shown by criticality analysis that the use of the three out of four storage configuration will assure that the K_{eff} will remain less than 0.95, including uncertainties, when fuel with a maximum enrichment of 4.25 weight percent U-235 and average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool.

The requirement for maintaining the spent fuel pool boron concentration greater than 500 ppm whenever fuel with average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool ensures that K_{eff} for the spent fuel pool will remain less than 0.95, including uncertainties, even if a fuel assembly is inadvertently inserted in the empty cell of the three out of four storage configuration.

References

1. USAR, Section 10.2.1.2
2. USAR, Section 14.5.1
3. USAR, Section 10.3.7

4.19 Auxiliary Building Crane Lifting Devices

Bases

The auxiliary building crane has been modified to conform with the single failure proof requirements of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980. The auxiliary building crane is designed to not allow a load drop as a result of any single failure. As the slings and special lifting devices are, by their nature, an integral part of the load bearing path, their surveillance is necessary to ensure against a load drop as a result of deficient rigging. Any load that weighs more than the combined weight of a single fuel assembly and its associated handling tool is considered a heavy load.

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
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 , 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.92, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 9, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 92

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 marginal lines indicating the area of change.

REMOVE	INSERT
TS-iii	TS-iii
TS-vi	TS-vi
TS-vii	TS-vii
TS-xi	TS-xi
TS.3.8-3	TS.3.8-3
Table TS.4.1-2B (page 1 of 2)	Table TS.4.1-2B (page 1 of 2)
Table TS.4.1-2B (page 2 of 2)	Table TS.4.1-2B (page 2 of 2)
----	TS.4.19-1
TS.5.6-1	TS.5.6-1
TS.5.6-2	TS.5.6-2
TS.5.6-3	----
B.3.8-1	B.3.8-1
B.3.8-2	B.3.8-2
B.3.8-3	----
----	B.4.19-1

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (continued)

3.8.B. Fuel Handling Operations

1. During fuel handling operations the following conditions shall be satisfied:

   a. Radiation levels in the spent fuel storage pool area shall be monitored continuously during fuel handling operations.

   b. Prior to fuel handling operations, fuel-handling cranes shall be load-tested for OPERABILITY of limit switches, interlocks and alarms.

   c. A minimum boron concentration of 1800 ppm shall be maintained in the spent fuel pool whenever a spent fuel cask containing fuel is located in the spent fuel pool.

2. If any of the conditions in 3.8.B.1, above, cannot be met, suspend all fuel handling operations and initiate the actions necessary to re-establish compliance with the requirements of 3.8.B.1.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
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. RCS Chemistry (Cl*,F*, O_2)	5/Week	
8. RCS Boron Concentration*(3)	2/Week (4)	9.2
9. RWST Boron Concentration	Weekly	
10. Boric Acid Tanks Boron Concentration	2/Week	
11. Caustic Standpipe NaOH Concentration	Monthly	6.4
12. Accumulator Boron Concentration	Monthly	6
13. Spent Fuel Pit Boron Concentration	Monthly (7)	9.5.5

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY	FSAR Section Reference
14. Secondary Coolant Gross Beta-Gamma activity	Weekly	
15. Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)	
16. Secondary Coolant Chemistry		
pH	5/week (6)	
pH Control Additive	5/week (6)	
Sodium	5/week (6)	

Notes:

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

2. To determine activity of corrosion products having a half-life greater than 30 minutes.

3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.

4. The maximum interval between analyses shall not exceed 5 days.

5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.

6. The maximum interval between analyses shall not exceed 3 days.

7. The minimum spent fuel pool boron concentration from Specification 3.8.B.1.b shall be verified by chemical analysis weekly while a spent fuel cask containing fuel is located in the spent fuel pool.

* See Specification 4.1.D

4.19 Auxiliary Building Crane Lifting Devices

Applicability

Applies to surveillance requirements for the auxiliary building crane special lifting devices and slings before handling heavy loads carried over safe shutdown equipment or spent fuel in the spent fuel pool.

Objective

To verify that special lifting devices and slings used in conjunction with the auxiliary building crane are operable prior to their use in supporting heavy loads over safe shutdown equipment or spent fuel in the spent fuel pool.

Specification

Slings and special lifting devices which will be used in supporting heavy loads from the auxiliary building crane shall be visually inspected and verified OPERABLE within 7 days prior to their use in handling heavy loads over safe shutdown equipment or spent fuel in the spent fuel pool.

5.6 FUEL HANDLING

A. Criticality Consideration

The new and spent fuel pit structures are designed to withstand the anticipated earthquake loadings as Class I (seismic) structures. The spent fuel pit has a stainless steel liner to ensure against loss of water (Reference 1).

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The design of the new fuel storage pit and racks (Reference 1) ensures a new fuel pit K_{eff} of less than or equal to 0.95, including uncertainties, even if unborated water were used to fill the pit. The new fuel rack configuration also ensures K_{eff} less than or equal to 0.98, including uncertainties, even if the new fuel racks were accidentally filled with a low density moderator which resulted in optimum low density moderation conditions. Fuel stored in the new fuel storage racks will have a maximum enrichment of 4.25 weight percent U-235.

The spent fuel storage rack design (Reference 1) and the limitations on the storage of low burnup fuel contained in Technical Specification Section 3.8.E ensure a spent fuel pool K_{eff} of less than or equal to 0.95, including uncertainties. The maximum enrichment of fuel to be stored in the spent fuel pool will be 4.25 weight percent U-235.

Fuel will not be inserted into a spent fuel cask in the pool, unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask. The criticality analysis for the TN-40 spent fuel storage cask was based on fresh fuel enriched to 3.85 weight percent U-235.

B. Spent Fuel Storage Structure

The spent fuel storage pool is enclosed with a reinforced concrete building having 12- to 18-inch thick walls and roof (Reference 1). The pool and pool enclosure are Class I (seismic) structures that afford protection against loss of integrity from postulated tornado missiles. The storage compartments and the fuel transfer canal are connected by fuel transfer slots that can be closed off with pneumatically sealed gates. The bottoms of the slots are above the tops of the active fuel in the fuel assemblies which will be stored vertically in specially constructed racks.

The spent fuel pool has a reinforced concrete bottom slab nearly 6 feet thick and has been designed to minimize loss of water due to a dropped cask accident. Piping to the pool is arranged so that failure of any pipe cannot drain the pool below the tops of the stored fuel assemblies.

C. Fuel Handling

The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after post-irradiation cooling. The system consists of the refueling cavity, the fuel transfer system, the spent fuel storage pit, and the spent fuel cask transfer system.

Major components of the fuel handling system are the manipulation crane, the spent fuel pool bridge, the auxiliary building crane, the fuel transfer system, the spent fuel storage racks, the spent fuel cask, and the rod cluster control changing fixture. The reactor vessel stud tensioner, the reactor vessel head lifting device, and the reactor internals lifting device are used for preparing the reactor for refueling and for assembling the reactor after refueling.

Upon arrival in the storage pit, spent fuel will be removed from the transfer system and placed, one assembly at a time, in storage racks using a long-handled manual tool suspended from the spent fuel pit bridge crane. After sufficient decay, the fuel will be loaded into storage casks for storage in the Independent Spent Fuel Storage Installation or into shipping casks for removal from the site. The casks will be handled by the auxiliary building crane.

Spent fuel casks will be handled by a single failure proof handling system meeting the requirements of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980. The auxiliary building crane has been upgraded to conform with the single failure proof requirements of Section 5.1.6 of NUREG-0612. The auxiliary building crane is designed to not allow a load drop as a result of any single failure. The improved reliability of the auxiliary building crane is achieved through increased factors of safety and through redundancy or duality in certain active components.

D. Spent Fuel Storage Capacity

The spent fuel storage facility is a two-compartment pool that, if completely filled with fuel storage racks, provides up to 1582 storage locations. The southeast corner of the small pool (pool no. 1) also serves as the cask lay down area. During times when the cask is being used, four racks are removed from the small pool. With the four storage racks in the southeast corner of pool 1 removed, a total of 1386 storage locations are provided. To allow insertion of a spent fuel cask, total storage is limited to 1386 assemblies, not including those assemblies which can be returned to the reactor.

Reference

1. USAR, Section 10.2

3.8 REFUELING AND FUEL HANDLING

Bases

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the precautions specified above, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during CORE ALTERATIONS that would result in a hazard to public health and safety (Reference 1). Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

Under rodded and unrodded conditions, the K_{eff} of the reactor must be less than or equal to 0.95 and the boron concentration must be greater than or equal to 2000 ppm. Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. 3.8.A.1.h allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

No movement of fuel in the reactor is permitted until the reactor has been subcritical for at least 100 hours to permit decay of the fission products in the fuel. The delay time is consistent with the fuel handling accident analysis (Reference 2).

Fuel will not be inserted into a spent fuel cask unless a minimum boron concentration of 1800 ppm is present. The 1800 ppm will ensure that k_{eff} for the spent fuel cask, including statistical uncertainties, will be less than or equal to 0.95 for all postulated arrangements of fuel within the cask.

The number of recently discharged assemblies in Pool No. 1 has been limited to 45 to provide assurance that in the event of loss of pool cooling capability, at least eight hours are available under worst case conditions to make repairs until the onset of boiling.

The Spent Fuel Pool Special Ventilation System (Reference 3) is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation System is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment.

3.8 REFUELING AND FUEL HANDLING

Bases continued

During movement of irradiated fuel assemblies or control rods, a water level of 23 feet is maintained to provide sufficient shielding.

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The water level may also be lowered below 20 feet for upper internals removal/replacement. The basis for these allowance(s) are (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching/unlatching and upper internals removal/replacement the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

The requirements for the storage of low burnup fuel in the spent fuel pool ensure that the spent fuel pool will remain subcritical during fuel storage. Fuel stored in the spent fuel pool will be limited to a maximum enrichment of 4.25 weight percent U-235. It has been shown by criticality analysis that the use of the three out of four storage configuration will assure that the K_{eff} will remain less than 0.95, including uncertainties, when fuel with a maximum enrichment of 4.25 weight percent U-235 and average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool.

The requirement for maintaining the spent fuel pool boron concentration greater than 500 ppm whenever fuel with average assembly burnup of less than 5,000 MWD/MTU is stored in the spent fuel pool ensures that K_{eff} for the spent fuel pool will remain less than 0.95, including uncertainties, even if a fuel assembly is inadvertently inserted in the empty cell of the three out of four storage configuration.

References

1. USAR, Section 10.2.1.2
2. USAR, Section 14.5.1
3. USAR, Section 10.3.7

4.19 Auxiliary Building Crane Lifting Devices

Bases

The auxiliary building crane has been modified to conform with the single failure proof requirements of Section 5.1.6 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", July 1980. The auxiliary building crane is designed to not allow a load drop as a result of any single failure. As the slings and special lifting devices are, by their nature, an integral part of the load bearing path, their surveillance is necessary to ensure against a load drop as a result of deficient rigging. Any load that weighs more than the combined weight of a single fuel assembly and its associated handling tool is considered a heavy load.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 99 AND 92 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 October 4, 1991, as supplemented by letter dated December 16, 1991, Northern States Power Company (NSP or the licensee) requested amendments 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. The proposed amendments would revise Technical Specification Section 3.8.B and its associated Bases to remove the restriction related to cask handling; add a new Section 4.19 and associated Bases which establish surveillance requirements for the Auxiliary Building crane lifting devices; and revise Section 5.6 to remove references to the spent fuel cask drop analysis and mitigation design features, and incorporate a new paragraph which states that spent fuel casks will be handled by a single-failure-proof handling system.

The amendments also make several changes of an administrative nature in Technical Specification Sections 3.8.B, 5.6 and in Table TS 4.1-2B in order to accommodate placement of spent fuel storage casks in the spent fuel pool, and to discuss the Bases for spent fuel boron requirements to maintain the boron concentration level, provide an action statement if boron concentration falls below required levels, and require a weekly verification of the boron concentration.

The licensee for the Prairie Island Nuclear Generating Plant, Units 1 and 2, intends to construct and operate a Dry Cask Independent Spent Fuel Storage Installation (ISFSI). The licensee submitted a proposal dated August 31, 1990, showing details of the proposed plan and provided further information in submittals dated September 26, and December 12, 1991. In submittals dated October 4, 1991 and February 3, 1992, the licensee provided information relating to upgrading of the auxiliary building crane, which is to be used in moving the cask containing the spent fuel from the spent fuel pool (SFP) to the ISFSI, so as to make the crane single-failure-proof. The licensee is making this change because the height of the TN-40 cask precludes the use of the impact limiter or crash pad, currently in the TS, as the means to limit

This Safety Evaluation (SE) is concerned primarily with the planned modifications of the auxiliary building crane so as to make it single-failure-proof together with those other portions of the heavy load handling system required to constitute a single-failure-proof system for movement of the TN-40 cask within the confines of the auxiliary building. It also addresses the administrative requirements necessary to maintain spent fuel pool boron concentration levels.

2.0 EVALUATION

2.1. AUXILIARY BUILDING CRANE

The licensee committed to upgrade the auxiliary building crane so as to have it comply with the criteria of Section 5.1.6, "Single-Failure-Proof Handling Systems," and Appendix C, "Modification of Existing Cranes" of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The licensee has contracted with Ederer Inc., to modify this crane in accordance with the provisions of Topical Report EDR-1(P)-A, "Ederer Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) Cranes." Details of the design were forwarded in a submittal dated February 3, 1992. As noted in the Ederer report, X-SAM hoists use three types of safety systems to protect against equipment malfunctions and operator error:

(1) Conventional Hoist Safety Systems, including upper and lower travel limits, overload sensing devices, hoist control protective features, and holding brake on the high speed shafting.

(2) The Hoist Integrated Protective System (HIPS) is used as a second line of defense. HIPS includes a special Emergency Drum Brake System, which acts on the wire rope drum; a Failure Detection System (FDS); and an Energy Absorbing Torque Limiter (EATL) in the drive train. The Failure Detection System actuates the Emergency Drum Brake System, which stops the wire rope drum in the event of a drive train discontinuity or component failure.

The EATL acts to retain the load in the event of two-blocking, overloading or load hangup by converting the hoist's kinetic energy to heat. The EATL to be built for the Prairie Island licensee will consist of 14 friction surfaces with a specified torque setting of approximately 130% of the main hoist design rated load. The capacity of the emergency drum brake to be designed for the Prairie Island Plant will be 130% of that required to hold the design rated load.

(3) The Balanced Dual Reeving System is the defense employed for X-SAM cranes against load sway and loss of load in the event of a single cable failure.

Other design details include:

(1) A maximum load motion of 1.5 feet, vertically, in the event of a drive train failure. Because of the features incorporated in the X-SAM crane design, the maximum kinetic energy to be absorbed from that motion would be that corresponding to a load free-fall of less than one inch. The licensee examined certain plant structures whose location would lie within 1.5 feet of the load during movement and ascertained either that damage to such structures would not prevent safe plant shutdown or that the structures could withstand the potential one inch free fall.

(2) No reverse bends in the wire rope except that between the wire rope drum and the first sheave in the load block. This is consistent with the guidelines in NUREG-0554 "Single-Failure-Proof Cranes for Nuclear Power Plants."

(3) Maximum fleet angle of 3.5 degrees in conformance with the criterion specified in NUREG-0554.

(4) The main hook does not have two independent paths. However, the single load path has an ultimate strength safety factor of 10:1 when lifting the maximum critical load of 125 tons. This is consistent with the guideline in Appendix C of NUREG-0612 for operating plants.

(5) Seismic Design - the licensee reported following the guidance of Section 2.5 "Seismic Design" of NUREG-0554, in qualifying the modified auxiliary building crane for seismic events.

The staff, in an SE dated August 26, 1983, found the Topical Report EDR-1(P), Revision 3, "Ederer Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) Cranes" to be suitable for reference in licensing applications. The SE required that Ederer publish, thereafter, accepted versions of the report and to append an -A, designating it be accepted as EDR-1(P)-A. The licensee cited this accepted version in its submittal. The licensee provided details specific to the modified design of the Prairie Island auxiliary building crane, as required by EDR-1(P)-A, some of which are discussed above. In addition, the licensee committed to modify the crane in accordance with the single-failure-proof criteria of NUREG-0612.

Based upon the licensee's commitments and the foregoing discussion, the staff finds the modified design of the Prairie Island auxiliary building crane to be in accordance with the guidelines of NUREG-0612 for single-failure-proof cranes.

2.2 Lifting System for Cask

In the February 4, 1992 submittal, the licensee reported a conceptual design for a lifting device, "Lift Beam Concept," together with the design criteria to be used for the lifting system. The design contains a crane hook adapter consisting of two parallel columns. The design also contains two cross beams and two lifting arms. The crane hook adapter attaches to the cross beams at one end and to the crane at the other end with 6-7/8 inch diameter pins. The cross beams are horizontal, parallel, and attached to lift arms through 6-7/8 inch diameter pins. The pins are used to swivel the lift arms into "in" and "out" positions so as to engage and disengage the bosses on the TN-40 cask by way of cutouts in the plates constituting the lift arms. Manual disengagement and locking devices just above the cross beams are used to move the arms and to lock them in place so as to hold the cask.

The licensee intends to use the stress design guidance factor of ANSI 14.6, "Special, Lifting Devices for Shipping Containers Weighing 10000 pounds (4500 kg) or More" in the design of the lifting system. The design is not truly single-failure-proof because a loss of one of the lift arms would permit the load to rotate about the other arm or to drop (the two cross beams and two crane hook adapter columns could be designed so that one beam or column could support the load in the event of a beam or column failure). However, the licensee intends to utilize the alternate method of providing stress design safety factors of 6 (i.e., 6 times the Maximum Critical Load) to be equal to or less than the material yield stress and 10 (i.e., 10 times the Maximum Critical Load) to be equal to or less than the material ultimate stress, as permitted by Section 5.1.6, "Single-Failure-Proof Handling Systems," of NUREG-0612. This criterion will apply to tensile, bearing, and shear stresses in the lift beam. In addition, the licensee is adding an additional factor of 5% (changing the factor 6 to 6.3 and 10 to 10.5) to account for dynamic loads. The licensee committed to justify use of the 5% factor for dynamic loads, prior to initial use.

There are additional safety features to prevent dropping the TN-40 casks. The lift arms, in engaging the TN-40 cask bosses, fit into notches in the bosses so that the TN-40 cask has to be jolted sufficiently while being carried to lift the cask from the notches. Even then, manual locking devices in the lift beam keep the arms from moving and prevent them from separating in such a way as to permit dropping the TN-40 cask. The lift beam may also be used to lift the TN-40 cask lid by means of slings attached to three eyes on both the lift beam and cask lid.

Prior to initial use of the lift beam, the licensee committed to provide an acceptable plan for its periodic testing to assure compliance with section 5.3, "Testing to Verify Continuing Compliance," of ANSI 14.6.

Based on the above, the licensee complies with the requirements of Paragraph (1)(a) "Special Lifting Devices" of Section 5.1.6, of NUREG-0612.

The staff finds the concept of the lift beam, together with the licensee's commitment to have the lifting system single-failure-proof and to have the design conform to the criteria of NUREG-0612 (which includes the criteria of ANSI-14.6), acceptable.

The staff finds the modification of the auxiliary building crane to be acceptable as a single-failure-proof crane capable of carrying maximum critical loads not exceeding 125 tons as discussed in Section 2.1 above. The lift beam conceptual design, as proposed by the licensee in attachment 3 of the submittal of February 3, 1992, is found to be acceptable as a single failure proof special lifting device only for handling the TN-40 cask in accordance with the discussion in Section 2.2 above and as noted in the test and drawing of attachment 3 to the February 3, 1992 submittal.

The licensee has committed to submit an acceptable plan for surveillance testing of the special lifting device (lift beam) to comply with section 5.3 of ANSI 14.6 prior to initial use. The licensee has also committed to justify use of the 5% factor for dynamic loads, also prior to initial use of the lift beam.

By letter dated November 1, 1991, the staff requested additional information related to the criticality aspects of the single-failure-proof crane modifications. The questions concerned the frequency of the fuel pool boron surveillance, and assumptions made in the criticality analysis. By letter dated December 16, 1991, the licensee provided the clarifying information. The licensee stated that, based on alarm setpoints, a weekly boron concentration surveillance is sufficient to ensure the 0.95 k_{eff} requirement during an inadvertent dilution event because the pool monitoring instrumentation would detect any spent fuel pool dilution event before significant dilution of the boron concentration would occur. The licensee also stated that the criticality analysis for the TN-40 spent fuel storage cask did not take credit for the burnup of the spent fuel stored in the cask.

Several changes are required to the TS in order to accommodate placement of the spent fuel storage casks in the spent fuel pool. These changes are administrative in nature or constitute additional restrictions not presently in the TS. Based on a review of the information provided by the licensee in the December 16, 1991, letter, the staff finds these changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulation, 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 requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change to the 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 (56 FR 50118). The supplementary information provided in a letter of December 16, 1991, was merely clarifying and did not change the scope of the action or the proposed finding of no significant hazards consideration. 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: N. Wagner

Date: July 9, 1992