50-282/306

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

December 7, 1998

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

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISED ADMINISTRATIVE CONTROLS (TAC NOS. M95130 AND M95131)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. DPR-42 and Amendment No. 132 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998.

The amendments revise TS 6.0, Administrative Controls, and the following TS sections affected by relocating, removing, and modifying requirements to TS 6.0: Table of Contents; TS 3.1, "Reactor Coolant System"; TS 4.0, "Surveillance Requirements"; TS 5.0, "Design Features"; and associated Bases. The removed or relocated requirements are adequately controlled by existing regulations other than 10 CFR 50.36 and the TS. In approving the proposed action, we have relied upon incorporating specific requirements as delineated in the enclosed safety evaluation into the Operational Quality Assurance Plan, Fire Protection Program, Security Plan, and Updated Safety Analysis Report. The changes associated with the Operational Quality Assurance Plan were approved on November 12, 1997. The amendments also add license conditions to Appendix B of the licenses. These license conditions were proposed by Northern States Power in its March 2, 1998, letter, as supplemented June 11 and October 30, 1998.

Because full implementation of these amendments may not take place until September 1, 1999, until full implementation Northern States Power should submit two sets of TS pages for any pages affected in future amendments by the issuance of these amendments. The TS pages should reflect the conditions before and after full implementation of these amendments so that the correct TS pages may be issued in any future amendments. The NRC also requests that you submit a letter informing the staff when these amendments are fully implemented.

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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

Carl F. Lyon, 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. 141 to DPR-42
2. Amendment No. 132 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

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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,

Carl F. Lyon

Carl F. Lyon, Project Manager
Project Directorate III-1
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Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 141 to DPR-42
2. Amendment No. 132 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

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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. 141
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 December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Facility Operating License No. DPR-42 is hereby amended to approve the relocation of certain Technical Specification requirements to licensee-controlled documents, as described in the licensee's application dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, and evaluated in the staff's safety evaluation attached to this amendment. This license is also hereby amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(5) of Facility Operating License No. DPR-42 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 141 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 141 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance, with full implementation of the attached technical specifications and License Condition 7 by September 1, 1999. License Condition 6 shall be implemented by the next USAR update, but no later than June 1, 1999. Implementation shall also include the relocation of Technical Specification requirements to the appropriate licensee-controlled documents as identified in the licensee's application dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, and evaluated in the staff's safety evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Carl F. Lyon

Carl F. Lyon, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments: 1. Changes to the Technical Specifications
2. Appendix B, page B-2

Date of Issuance: December 7, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 141

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-ii	TS-ii
TS-v	TS-v
TS-viii	TS-viii
TS-ix	TS-ix
TS-x	TS-x
TS-xi	--
TS-xii	--
TS-xiii	--
TS.3.1-10	TS.3.1-10
TS.3.1-11	--
Table TS.4-1-2B (Page 1)	Table TS.4-1-2B (Page 1)
TS.4.4-3	TS.4.4-3
TS.4.6-1	TS.4.6-1
TS.5.1-1	TS.5.1-1
TS.5.1-2	TS.5.1-2
TS.6-1-1 through 6.1-4	--
Table TS.6.1-1	--
TS.6.2-1 through TS.6.2-7	--
TS.6.3-1	--
TS.6.5-1 through TS.6.5-8	--
TS.6.6-1 through TS.6.6-2	--
TS.6.7-1 through TS.6.7-6	--
Table TS.6.7-1	--
--	TS.6.0-1 through TS.6.0-17
B.3.1-8	B.3.1-8
B.3.1-9	--
B.4.4-3	B.4.4-3

Revise Appendix B, Additional Conditions, by removing the pages identified below and inserting the attached pages.

--	B-2

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TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TECHNICAL SPECIFICATIONS

LIST OF TABLES

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

3.1.D. MAXIMUM COOLANT ACTIVITY

1. The specific activity of the primary coolant (except as specified in 3.1.D.2 and 3 below) shall be limited to:

   a. Less than or equal to 1.0 microcuries per gram DOSE EQUIVALENT I-131, and

   b. Less than or equal to 100/$\bar{E}$ microcuries per gram of gross radioactivity.

2. If a reactor is critical or the reactor coolant system average temperature is greater than or equal to 500°F:

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure TS.3.1-3, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

   b. With the specific activity of the primary coolant greater than 100/$\bar{E}$ microcurie per gram, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

3. If a reactor is at or above COLD SHUTDOWN, with the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.1-2B until the specific activity of the primary coolant is restored to within its limits.

Next pages are Figure TS.3.1-3 and TS.3.1-12.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	TEST	FREQUENCY
1.	RCS Gross Activity Determination	5/week
2.	RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)
3.	RCS Radiochemistry Ē 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/Ē uCi/gram (at or above cold shutdown), and b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)
5.	RCS Radiochemistry (2)	Monthly
6.	RCS Tritium Activity	Weekly
7.	Deleted	
8.	RCS Boron Concentration*(3)	2/Week (4)
9.	RWST Boron Concentration	Weekly
10.	Boric Acid Tanks Boron Concentration	2/Week
11.	Caustic Standpipe NaOH Concentration	Monthly
12.	Accumulator Boron Concentration	Monthly
13.	Spent Fuel Pit Boron Concentration	Weekly

*Required at all times.

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate ($\pm$10%). The results of the test shall show the air distribution is uniform within $\pm$20%.

C. <u>Containment Vacuum Breakers</u>

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containmant vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested in accordance with the requirements of Specification 4.4.A.3.

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power sources and equipment are OPERABLE.

Specification

The following tests and surveillance shall be performed:

A. Diesel Generators

1. At least once each month, for each diesel generator:

   a. Verify the fuel level in the day tank.

   b. Verify the fuel level in the fuel storage tank.

   c. Deleted

   d. Verify the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

   e. Verify the diesel generator can start and gradually accelerate. Verify the generator voltage and frequency can be adjusted to 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz. Subsequently, manually sychronize the generator, gradually load to at least 1650 kW (Unit 2: 5100 kW to 5300 KW), and operate for at least 60 minutes. This test should be conducted in consideration of the manufacturer's recommendations regarding engine prelube, warm-up, loading and shutdown procedures where possible.

5.0 DESIGN FEATURES

5.1 SITE

The Prairie Island Nuclear Generating Plant is located on property owned by Northern States Power (NSP) Company at a site on the west bank of the Mississippi River, approximately 6 miles northwest of the city of Red Wing, Minnesota. The minimum distance from the center line of either reactor to the site exclusion boundary is 715 meters, and the low population zone distance is 1-1/2 miles. The nearest population center of 25,000 or more people is South Saint Paul. These site characteristics comply with definitions in 10CFR100 (Reference 1).

The U.S. Army Corp of Engineers controls the land within the exclusion area that is not owned by NSP. The Corps has made an agreement with NSP to prevent residential construction on this land for the life of the plant (Reference 2).

These specifications use atmospheric diffusion factors based on the NRC staff evaluations. Its evaluation of accidental airborne releases is based on a relative concentration of 9.8×10^{-4} seconds per cubic meter at the site boundary. Its evaluation of routine releases is based on a relative concentration of 1.5×10^{-5} seconds per cubic meter (Reference 3).

The flood of record in 1965 produced a water surface elevation of +688 feet MSL at the site. The calculated probable maximum flood (PMF) level is +703.6 feet mean sea level (MSL), and the estimated wave runup could reach +706.7 feet MSL. (See Section 2.4.2 of this report.) Plant grade level is +695 feet MSL.

Flood protection structures have been provided. The two turbine support facilities, the common auxiliary building, and the two shield buildings have been physically connected by a concrete flood wall, most of the length of which constitutes the concrete foundation walls for the various buildings. The top of this wall supports the metal siding for the buildings at about elevation +705 feet MSL. Fourteen doors through the flood wall, or into the various buildings (including the separate screen house), are provided with receivers for the erection of flood protection panels to prevent flood water from reaching safety related facilities.

The cooling water pumps in the screenhouse are designed to operate up to a flood level of +695 feet MSL without flood protection measures, and up to a level of +707 feet MSL with the erection of flood protection panels. The main transformer foundation is at +695 feet MSL. The transformer will function to a flood level of +698 feet MSL.

5.1 SITE (continued)

The plant is designed for a design basis earthquake having a horizontal ground acceleration of 0.12g and an operational basis earthquake having a horizontal ground acceleration of 0.06g.

References

1. USAR, Section 2.2.1
2. USAR, Section 3.4.5
3. SER, Sections 2.3.4 and 2.3.5

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

A. The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

B. The shift supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active senior reactor operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator license shall be designated to assume the control room command function.

6.2 Organization

A. Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

1. Lines of authority, responsibility and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Safety Analysis Report.

2. The plant manager shall report to the corporate officer specified in 6.2.A.3, shall be responsible for overall safe operation of the plant, and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

3. A corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Plant Staff

The plant staff organization shall include the following:

1. An operator to perform non-licensed duties shall be assigned to each reactor containing fuel and one additional operator to perform non-licensed duties shall be assigned when either or both reactors are operating in MODES 1, 2, 3, or 4.

2. At least one licensed operator shall be present in the control room for each reactor containing fuel. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed senior reactor operator shall be present in the control room.

B. Plant Staff (continued)

3. Shift crew composition may be less than the minimum requirement of 10CFR50.54(m)(2)(i) and 6.2.B.1 and 6.2.B.7 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).

   The procedures shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviations from the working hour guidelines shall be authorized in advance by the Plant Manager or designee in accordance with approved administrative procedures and with documentation of the basis for granting the deviation.

   Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or designee, to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

6. The operations manager or assistant operations manager shall hold an SRO license.

7. The shift technical advisor (STA) shall provide advisory technical support to the shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. Personnel performing the function of the STA shall be assigned to the shift crew when a unit is in MODE 1, 2, 3, or 4.

6.3 Plant Staff Qualifications

Each member of the plant staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975 except for (1) personnel who perform the function of shift technical advisor shall hold an SRO license and have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (2) the operations manager who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.2.B.6.

6.4 Procedures

Written procedures shall be established, implemented, and maintained covering the following activities:

A. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

B. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

C. Quality control for effluent and environmental monitoring;

D. Fire protection program implementation; and

E. All programs specified in Specification 6.5.

6.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

A. Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Monitoring, and Radioactive Effluent Reports required by Specification 6.6.B and Specification 6.6.C.

Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   b. a determination that the change(s) maintain the levels of radioactive effluent control required by 10CFR20.1302, 40CFR190, 10CFR50.36a, and 10CFR50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations;

2. Shall become effective after approval by a member of plant management designated by the Plant Manager.

3. Shall be submitted to the NRC in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed. The date (i.e., month and year) the change was implemented shall be indicated.

B. Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The systems include portions of Residual Heat Removal, Safety Injection, and Containment Spray Systems. The program shall include the following:

B. Primary Coolant Sources Outside Containment (continued)

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

C. Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel;

2. Procedures for sampling and analysis; and

3. Provisions for maintenance of sampling and analysis equipment.

D. Radioactive Effluent Controls Program

This program conforms to 10CFR50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable.

This program shall allocate releases equally to each unit. The liquid radwaste treatment system, waste gas treatment system, containment purge release vent, and spent fuel pool vent are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. Therefore, all releases will be allocated equally in determining conformance to the design objectives of 10CFR50, Appendix I.

The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

2. Limitation on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to Appendix B to 10CFR20.1 - 20.601, Table II, Column 2;

D. Radioactive Effluent Controls Program (continued)

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10CFR20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10CFR50, Appendix I;

5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least monthly;

6. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of one month from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with Appendix B to 10CFR20.1 - 20.601, Table II, Column 1;

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10CFR50, Appendix I;

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10CFR50, Appendix I; and

10. Limitation on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40CFR190.

E. Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4 cyclic and transient occurrences to ensure that components are maintained within the design limits.

F. (Reserved)

G. (Reserved)

H. (Reserved)

I. (Reserved)

J. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentration of oxygen in the waste gas holdup system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria;

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 78,800 curies of noble gases (considered as dose equivalent Xe-133); and

3. A surveillance program to ensure that the quantity of radioactivity contained in each of the following tanks shall be limited to 10 curies, excluding tritium and dissolved or entrained noble gases:

   Condensate storage tanks
   Outside temporary tanks

4. The provisions of TS 4.0 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

K. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment. Acceptability of new fuel oil shall be determined prior to addition to the safeguards storage tanks. Testing of diesel fuel oil stored in safeguards storage tanks shall be performed at least every 31 days. The provisions of TS 4.0 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

L. Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   a. a change in the Technical Specifications incorporated in the license; or

   b. a change to the USAR or Bases that involves an unreviewed safety question as defined in 10CFR50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

4. Proposed changes that meet the criteria of Specifications 6.5.L.2.a and b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with USAR updates.

M. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.

Leakage Rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria

M. <u>Containment Leakage Rate Testing Program</u> (continued)

are $\leq 0.60\ L_a$ for all components subject to Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 46 psig

2) For each door intergasket test, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

The provisions of 4.0.A do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. The Containment Leakage Rate Testing Program stipulates acceptable extension of test intervals.

The provisions of 4.0.B (except that the allowed surveillance intervals are defined by the Containment Leakage Rate Testing Program) are applicable to the Containment Leakage Rate Testing Program.

6.6 Reporting Requirements

The following reports shall be submitted in accordance with 10CFR50.4

A. Occupational Exposure Report

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10CFR20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. This report shall be submitted by April 30 of each year.

B. Annual Radiological Environmental Monitoring Report

The Annual Radiological Environmental Monitoring Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10CFR50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiation Environmental Monitoring Reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

C. Radioactive Effluent Report

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR50.36a and 10CFR50, Appendix I, Section IV.B.1.

D. Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

E. Core Operating Limits Report (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   a. Heat Flux Hot Channel Factor Limit (F_Q^{RTP}), Nuclear Enthalpy

      Rise Hot Channel Factor Limit ($F\Delta_H^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

   b. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

   c. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

   d. Reactor Coolant System Flow Limit (Specification 3.10.J)

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

   NSPNAD-8102-A, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units"(latest approved version)

E. Core Operating Limits Report (COLR) (continued)

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990

XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors PhaseII", May, 1981

WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO$_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993)

NSPNAD-93003-A, "Transient Power Distribution Methodology",(latest approved version)

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

4. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

F. Pressure and Temperature Limit Report

1. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following Technical Specification sections: 3.1.A.1.c(4), 3.1.A.2.c(2), 3.1.A.2.c(3), 3.1.B.1.a, 3.3.A.3, 3.3.A.4, 3.3.A.5, and Table 4.1-1C.

2. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and

Cooldown Limit Curves" (Includes any exemption granted by NRC to ASME Code Case N-514)

F. <u>Pressure and Temperature Limit Report</u> (continued)

3. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties will be submitted to the NRC prior to issuance of an updated PTLR.

6.7 High Radiation Area

A. Pursuant to 10CFR20, paragraph 20.1601(c), in lieu of the requirements of 10CFR20.1601, each high radiation area, as defined in 10CFR20, in which the intensity of radiation is greater than 100 mrem/hr but less than or equal to 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., health physics technicians) or personnel

continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates less than or equal to 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A radiation monitoring device that continuously indicates the radiation dose rate in the area.
2. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.
3. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection manager.

B. In addition to the requirements of Specification 6.7.A above, areas with radiation levels greater than 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV or transmitting radiation monitoring device) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

6.7 High Radiation Area (continued)

C. For individual high radiation areas with radiation levels of greater than 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for puposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

3.1 REACTOR COOLANT SYSTEM

Bases continued

D. Maximum Coolant Activity

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Prairie Island site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

Specification 3.1.D.2, permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure TS.3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure TS.3.1-3 should be minimized since the activity levels allowed by Figure TS.3.1-3 increase the 2 hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing RCS temperature to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements in Table TS.4.1-2B provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

Next page is B.3.1-10.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the Shield Building Ventilation System (Reference 5). Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the Auxiliary Building Special Ventilation System. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of-coolant accident was initially evaluated by the AEC staff (Reference 3) assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest SITE BOUNDARY and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment in-leakage (Reference 6).

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Amendment Number	Additional Condition	Implementation Date
141	6. Relocate current Technical Specification 3.1.E, Maximum Reactor Oxygen, Chloride and Fluoride Concentration, Technical Specification 5.1 flood shutdown requirements to the USAR.	By the next USAR update, but no later than June 1, 1999
141	7. Relocate current Technical Specification 4.6.A.1.c, Diesel Fuel Oil Testing, requirements to the Diesel Fuel Oil Testing Program.	By September 1, 1999

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. 132
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 December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Facility Operating License No. DPR-60 is hereby amended to approve the relocation of certain Technical Specification requirements to licensee-controlled documents, as described in the licensee's application dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, and evaluated in the staff's safety evaluation attached to this amendment. This license is also amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C.(5) of Facility Operating License No. DPR-60 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 132 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 132 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance, with full implementation of the attached technical specifications and License Condition 7 by September 1, 1999. License Condition 6 shall be implemented by the next USAR update, but no later than June 1, 1999. Implementation shall also include the relocation of Technical Specification requirements to the appropriate licensee-controlled documents as identified in the licensee's application dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, and evaluated in the staff's safety evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Carl F. Lyon

Carl F. Lyon, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments: 1. Changes to the Technical Specifications
2. Appendix B page B-2

Date of Issuance: December 7, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 132

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-ii	TS-ii
TS-v	TS-v
TS-viii	TS-viii
TS-ix	TS-ix
TS-x	TS-x
TS-xi	--
TS-xii	--
TS-xiii	--
TS.3.1-10	TS.3.1-10
TS.3.1-11	--
Table TS.4-1-2B (Page 1)	Table TS.4-1-2B (Page 1)
TS.4.4-3	TS.4.4-3
TS.4.6-1	TS.4.6-1
TS.5.1-1	TS.5.1-1
TS.5.1-2	TS.5.1-2
TS.6-1-1 through 6.1-4	--
Table TS.6.1-1	--
TS.6.2-1 through TS.6.2-7	--
TS.6.3-1	--
TS.6.5-1 through TS.6.5-8	--
TS.6.6-1 through TS.6.6-2	--
TS.6.7-1 through TS.6.7-6	--
Table TS.6.7-1	--
--	TS.6.0-1 through TS.6.0-17
B.3.1-8	B.3.1-8
B.3.1-9	--
B.4.4-3	B.4.4-3

Revise Appendix B, Additional Conditions, by removing the pages identified below and inserting the attached pages.

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TECHNICAL SPECIFICATIONS

LIST OF TABLES

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

3.1.D. MAXIMUM COOLANT ACTIVITY

1. The specific activity of the primary coolant (except as specified in 3.1.D.2 and 3 below) shall be limited to:

   a. Less than or equal to 1.0 microcuries per gram DOSE EQUIVALENT I-131, and

   b. Less than or equal to 100/Ē microcuries per gram of gross radioactivity.

2. If a reactor is critical or the reactor coolant system average temperature is greater than or equal to 500°F:

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure TS.3.1-3, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

   b. With the specific activity of the primary coolant greater than 100/Ē microcurie per gram, the reactor shall be shutdown and reactor coolant system average temperature cooled to below 500°F within 6 hours.

3. If a reactor is at or above COLD SHUTDOWN, with the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.1-2B until the specific activity of the primary coolant is restored to within its limits.

Next pages are Figure TS.3.1-3 and TS.3.1-12.

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

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

*Required at all times.

b. Cold DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing that could affect the HEPA bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing that could affect the charcoal adsorber bank bypass leakage.

d. Each circuit shall be operated with the heaters on at least 10 hours every month.

5. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at rated flow rate ($\pm$10%). The results of the test shall show the air distribution is uniform within $\pm$20%.

C. Containment Vacuum Breakers

The air-operated valve in each vent line shall be tested at quarterly intervals to demonstrate that a simulated containmant vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check valves as well as the butterfly valves will be leak-tested in accordance with the requirements of Specification 4.4.A.3.

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power sources and equipment are OPERABLE.

Specification

The following tests and surveillance shall be performed:

A. Diesel Generators

1. At least once each month, for each diesel generator:

   a. Verify the fuel level in the day tank.

   b. Verify the fuel level in the fuel storage tank.

   c. Deleted

   d. Verify the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

   e. Verify the diesel generator can start and gradually accelerate. Verify the generator voltage and frequency can be adjusted to 4160 ± 420 volts and 60 ± 1.2 Hz. Subsequently, manually sychronize the generator, gradually load to at least 1650 kW (Unit 2: 5100 kW to 5300 KW), and operate for at least 60 minutes. This test should be conducted in consideration of the manufacturer's recommendations regarding engine prelube, warm-up, loading and shutdown procedures where possible.

5.0 DESIGN FEATURES

5.1 SITE

The Prairie Island Nuclear Generating Plant is located on property owned by Northern States Power (NSP) Company at a site on the west bank of the Mississippi River, approximately 6 miles northwest of the city of Red Wing, Minnesota. The minimum distance from the center line of either reactor to the site exclusion boundary is 715 meters, and the low population zone distance is 1-1/2 miles. The nearest population center of 25,000 or more people is South Saint Paul. These site characteristics comply with definitions in 10CFR100 (Reference 1).

The U.S. Army Corp of Engineers controls the land within the exclusion area that is not owned by NSP. The Corps has made an agreement with NSP to prevent residential construction on this land for the life of the plant (Reference 2).

These specifications use atmospheric diffusion factors based on the NRC staff evaluations. Its evaluation of accidental airborne releases is based on a relative concentration of 9.8×10^{-4} seconds per cubic meter at the site boundary. Its evaluation of routine releases is based on a relative concentration of 1.5×10^{-5} seconds per cubic meter (Reference 3).

The flood of record in 1965 produced a water surface elevation of +688 feet MSL at the site. The calculated probable maximum flood (PMF) level is +703.6 feet mean sea level (MSL), and the estimated wave runup could reach +706.7 feet MSL. (See Section 2.4.2 of this report.) Plant grade level is +695 feet MSL.

Flood protection structures have been provided. The two turbine support facilities, the common auxiliary building, and the two shield buildings have been physically connected by a concrete flood wall, most of the length of which constitutes the concrete foundation walls for the various buildings. The top of this wall supports the metal siding for the buildings at about elevation +705 feet MSL. Fourteen doors through the flood wall, or into the various buildings (including the separate screen house), are provided with receivers for the erection of flood protection panels to prevent flood water from reaching safety related facilities.

The cooling water pumps in the screenhouse are designed to operate up to a flood level of +695 feet MSL without flood protection measures, and up to a level of +707 feet MSL with the erection of flood protection panels. The main transformer foundation is at +695 feet MSL. The transformer will function to a flood level of +698 feet MSL.

5.1 SITE (continued)

The plant is designed for a design basis earthquake having a horizontal ground acceleration of 0.12g and an operational basis earthquake having a horizontal ground acceleration of 0.06g.

References

1. USAR, Section 2.2.1
2. USAR, Section 3.4.5
3. SER, Sections 2.3.4 and 2.3.5

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

A. The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

B. The shift supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active senior reactor operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or reactor operator license shall be designated to assume the control room command function.

6.2 Organization

A. Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

1. Lines of authority, responsibility and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Safety Analysis Report.

2. The plant manager shall report to the corporate officer specified in 6.2.A.3, shall be responsible for overall safe operation of the plant, and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

3. A corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

B. Plant Staff

The plant staff organization shall include the following:

1. An operator to perform non-licensed duties shall be assigned to each reactor containing fuel and one additional operator to perform non-licensed duties shall be assigned when either or both reactors are operating in MODES 1, 2, 3, or 4.

2. At least one licensed operator shall be present in the control room for each reactor containing fuel. In addition, while either unit is in MODE 1, 2, 3, or 4, at least one licensed senior reactor operator shall be present in the control room.

B. Plant Staff (continued)

3. Shift crew composition may be less than the minimum requirement of 10CFR50.54(m)(2)(i) and 6.2.B.1 and 6.2.B.7 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).

   The procedures shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviations from the working hour guidelines shall be authorized in advance by the Plant Manager or designee in accordance with approved administrative procedures and with documentation of the basis for granting the deviation.

   Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or designee, to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

6. The operations manager or assistant operations manager shall hold an SRO license.

7. The shift technical advisor (STA) shall provide advisory technical support to the shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. Personnel performing the function of the STA shall be assigned to the shift crew when a unit is in MODE 1, 2, 3, or 4.

6.3 Plant Staff Qualifications

Each member of the plant staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 1, September 1975 except for (1) personnel who perform the function of shift technical advisor shall hold an SRO license and have a bachelors degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and (2) the operations manager who shall meet the requirements of ANSI N18.1-1971, except that NRC license requirements are as specified in Specification 6.2.B.6.

6.4 Procedures

Written procedures shall be established, implemented, and maintained covering the following activities:

A. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

B. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

C. Quality control for effluent and environmental monitoring;

D. Fire protection program implementation; and

E. All programs specified in Specification 6.5.

6.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

A. Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Monitoring, and Radioactive Effluent Reports required by Specification 6.6.B and Specification 6.6.C.

Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   b. a determination that the change(s) maintain the levels of radioactive effluent control required by 10CFR20.1302, 40CFR190, 10CFR50.36a, and 10CFR50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations;

2. Shall become effective after approval by a member of plant management designated by the Plant Manager.

3. Shall be submitted to the NRC in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed. The date (i.e., month and year) the change was implemented shall be indicated.

B. Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The systems include portions of Residual Heat Removal, Safety Injection, and Containment Spray Systems. The program shall include the following:

B. <u>Primary Coolant Sources Outside Containment</u> (continued)

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

C. <u>Post Accident Sampling</u>

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel;

2. Procedures for sampling and analysis; and

3. Provisions for maintenance of sampling and analysis equipment.

D. <u>Radioactive Effluent Controls Program</u>

This program conforms to 10CFR50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable.

This program shall allocate releases equally to each unit. The liquid radwaste treatment system, waste gas treatment system, containment purge release vent, and spent fuel pool vent are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. Therefore, all releases will be allocated equally in determining conformance to the design objectives of 10CFR50, Appendix I.

The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

2. Limitation on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to Appendix B to 10CFR20.1 - 20.601, Table II, Column 2;

D. Radioactive Effluent Controls Program (continued)

3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10CFR20.1302 and with the methodology and parameters in the ODCM;

4. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10CFR50, Appendix I;

5. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least monthly;

6. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of one month from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with Appendix B to 10CFR20.1 - 20.601, Table II, Column 1;

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10CFR50, Appendix I;

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10CFR50, Appendix I; and

10. Limitation on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40CFR190.

E. Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4 cyclic and transient occurrences to ensure that components are maintained within the design limits.

F. (Reserved)

G. (Reserved)

H. (Reserved)

I. (Reserved)

J. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas holdup system, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentration of oxygen in the waste gas holdup system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria;

2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than or equal to 78,800 curies of noble gases (considered as dose equivalent Xe-133); and

3. A surveillance program to ensure that the quantity of radioactivity contained in each of the following tanks shall be limited to 10 curies, excluding tritium and dissolved or entrained noble gases:

   Condensate storage tanks
   Outside temporary tanks

4. The provisions of TS 4.0 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

K. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment. Acceptability of new fuel oil shall be determined prior to addition to the safeguards storage tanks. Testing of diesel fuel oil stored in safeguards storage tanks shall be performed at least every 31 days. The provisions of TS 4.0 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies.

L. Technical Specifications Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the Technical Specifications shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   a. a change in the Technical Specifications incorporated in the license; or

   b. a change to the USAR or Bases that involves an unreviewed safety question as defined in 10CFR50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

4. Proposed changes that meet the criteria of Specifications 6.5.L.2.a and b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with USAR updates.

M. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a, of 46 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.25% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.1% of primary containment air weight per day at pressure P_a. For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.01% of primary containment air weight per day at pressure P_a.

Leakage Rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to unit startup, following testing in accordance with the program, the combined leakage rate acceptance criteria

M. Containment Leakage Rate Testing Program (continued)

are ≤ 0.60 L_a for all components subject to Type B and Type C tests and ≤0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥46 psig

2) For each door intergasket test, leakage rate is ≤ 0.01 L_a when pressurized to ≥10 psig.

The provisions of 4.0.A do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. The Containment Leakage Rate Testing Program stipulates acceptable extension of test intervals.

The provisions of 4.0.B (except that the allowed surveillance intervals are defined by the Containment Leakage Rate Testing Program) are applicable to the Containment Leakage Rate Testing Program.

6.6 Reporting Requirements

The following reports shall be submitted in accordance with 10CFR50.4

A. Occupational Exposure Report

A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10CFR20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. This report shall be submitted by April 30 of each year.

B. Annual Radiological Environmental Monitoring Report

The Annual Radiological Environmental Monitoring Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10CFR50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiation Environmental Monitoring Reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations keyed to a table giving distances and directions from the reactor site; and the results of licensees participation in the Interlaboratory Comparison Program defined in the ODCM.

C. Radioactive Effluent Report

The Radioactive Effluent Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR50.36a and 10CFR50, Appendix I, Section IV.B.1.

D. Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

E. Core Operating Limits Report (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   a. Heat Flux Hot Channel Factor Limit (F_Q^{RTP}), Nuclear Enthalpy

      Rise Hot Channel Factor Limit ($F\Delta_H^{RTP}$), PFDH, K(Z) and V(Z) (Specifications 3.10.B.1, 3.10.B.2 and 3.10.B.3)

   b. Axial Flux Difference Limits and Target Band (Specifications 3.10.B.4 through 3.10.B.9)

   c. Shutdown and Control Bank Insertion Limits (Specification 3.10.D)

   d. Reactor Coolant System Flow Limit (Specification 3.10.J)

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to PI Units" (latest approved version)

   NSPNAD-8102-A, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units"(latest approved version)

E. Core Operating Limits Report (COLR) (continued)

WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology", July, 1985

WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code", August, 1985

WCAP-10924-P-A, "Westinghouse Large-Break LOCA Best-Estimate Methodology", December, 1988

WCAP-10924-P-A, Volume 1, Addendum 4, "Westinghouse Large Break LOCA Best Estimate Methodology", August, 1990

XN-NF-77-57 (A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors PhaseII", May, 1981

WCAP-13677, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support $ZIRLO_{TM}$ Cladding Options", April 1993 (approved by NRC SE dated November 26, 1993)

NSPNAD-93003-A, "Transient Power Distribution Methodology",(latest approved version)

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

4. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

F. Pressure and Temperature Limit Report

1. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following Technical Specification sections: 3.1.A.1.c(4), 3.1.A.2.c(2), 3.1.A.2.c(3), 3.1.B.1.a, 3.3.A.3, 3.3.A.4, 3.3.A.5, and Table 4.1-1C.

2. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Includes any exemption granted by NRC to ASME Code Case N-514)

F. <u>Pressure and Temperature Limit Report</u> (continued)

3. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties will be submitted to the NRC prior to issuance of an updated PTLR.

6.7 High Radiation Area

A. Pursuant to 10CFR20, paragraph 20.1601(c), in lieu of the requirements of 10CFR20.1601, each high radiation area, as defined in 10CFR20, in which the intensity of radiation is greater than 100 mrem/hr but less than or equal to 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., health physics technicians) or personnel

continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates less than or equal to 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

2. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

3. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection manager.

B. In addition to the requirements of Specification 6.7.A above, areas with radiation levels greater than 1000 mrem/hr shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV or transmitting radiation monitoring device) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

6.7 High Radiation Area (continued)

C. For individual high radiation areas with radiation levels of greater than 1000 mrem/hr, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for puposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

3.1 REACTOR COOLANT SYSTEM

Bases continued

D. Maximum Coolant Activity

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Prairie Island site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

Specification 3.1.D.2, permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure TS.3.1-3, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure TS.3.1-3 should be minimized since the activity levels allowed by Figure TS.3.1-3 increase the 2 hour thyroid dose at the SITE BOUNDARY by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing RCS temperature to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements in Table TS.4.1-2B provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action.

Next page is B.3.1-10.

4.4 CONTAINMENT SYSTEM TESTS

Bases continued

Several penetrations of the containment vessel and the shield building could, in the event of leakage past their isolation valves, result in leakage being conveyed across the annulus by the penetrations themselves, thus bypassing the function of the Shield Building Ventilation System (Reference 5). Such leakage is estimated not to exceed .025% per day. A special zone of the auxiliary building has minimum-leakage construction and controlled access, and is designated as a special ventilation zone where such leakage would be collected by either of two redundant trains of the Auxiliary Building Special Ventilation System. This system, when activated, will supplant the normal ventilation and draw a vacuum throughout the zone such that all outleakage will be through particulate and charcoal filters which exhaust to the shield building exhaust stack.

The design basis loss-of-coolant accident was initially evaluated by the AEC staff (Reference 3) assuming primary containment leak rate of 0.5% per day at the peak accident pressure. Another conservative assumption in the calculation is that primary containment leakage directly to the ABSVZ is 0.1% per day and leakage directly to the environs is 0.01% per day. The resulting two-hour doses at the nearest SITE BOUNDARY and 30-day doses at the low population zone radius of 1½ miles are less than guidelines presented in 10CFR100.

Initial leakage testing of the shield building and the ABSV resulted in a greater inleakage than the design basis. The staff has reevaluated doses for these higher inleakage rates and found that for a primary containment leak rate of 0.25% per day at peak accident pressure, the offsite doses are about the same as those initially calculated for higher primary containment leakage and lower secondary containment in-leakage (Reference 6).

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Amendment Number	Additional Condition	Implementation Date
132	6. Relocate current Technical Specification 3.1.E, Maximum Reactor Oxygen, Chloride and Fluoride Concentration, Technical specification 5.1 flood shutdown requirements to the USAR (TRM).	By the next USAR update, but no later than June 1, 1999
132	7. Relocate current Technical Specification 4.6.A.1.c, Diesel Fuel Oil Testing, requirements to the Diesel Fuel Oil Testing Program.	By September 1, 1999

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141

TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 132 TO FACILITY OPERATION LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated December 14, 1995, as supplemented on November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, 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, Units 1 and 2, respectively. The proposed changes would modify Section 6.0 by removing or relocating requirements that are adequately controlled by existing regulations other than 10 CFR 50.36 and the TS. Guidance on the proposed changes was developed by the NRC and provided in the Standard Technical Specifications (STS) for Westinghouse Plants, NUREG-1431, and Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," issued on December 12, 1995.

The November 25, 1996, April 10, September 4, and December 29, 1997, January 8, March 2, June 11, August 12, and October 30, 1998, submittals provided additional clarifying information, revised implementation dates, and updated TS pages. This information was within the scope of the original *Federal Register* notice and did not change the staff's initial proposed no significant hazards considerations determination.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in Title 10, *Code of Federal Regulations* (CFR), Section 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission amended 10 CFR 50.36 (60 FR 36593, July 19, 1995), and codified four criteria to be used in determining whether a particular matter is required to be included in an LCO, as follows: (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; or (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. LCOs and related requirements that fall within or satisfy any of the criteria in the regulation must be retained in the TS, while those requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. While the criteria specifically apply to LCOs, in adopting the revision to the rule the Commission noted that the staff had used the intent of these criteria to identify the optimum set of administrative controls in the TS (60 FR 36957).

The regulation at 10 CFR 50.36 states that Administrative Controls "are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure safe operation of the facility in a safe manner." The specific content of the administrative controls section of the TS is, therefore, that information that the Commission deems essential for the safe operation of the facility that is not already adequately covered by other regulations. Accordingly, the staff has determined that requirements that are not specifically required under 10 CFR 50.36(c)(5), and that are not otherwise necessary for operation of the facility in a safe manner, can be removed from administrative controls.

3.0 EVALUATION

The following discussions detail the staff's conclusions regarding the removal or relocation of selected administrative controls from the Prairie Island Nuclear Generating Plant TS. The changes were reviewed in accordance with the guidance provided in, or planned for, the STS, NUREG-1431. In addition, these changes were reviewed in accordance with the guidance provided in Administrative Letter 95-06.

License amendment requests should describe the relocation of each selected requirement to a particular licensee-controlled document or program (e.g., the final safety analysis report (FSAR) or the quality assurance (QA) plan). The description should also address the submittal of the revised documents to the NRC in accordance with the applicable regulation (e.g., 10 CFR 50.71(e)). In the amendment request, the licensee should clearly describe the program it will use to control changes to relocated requirements (e.g., 10 CFR 50.59 or 50.54(q)). Control of the relocated requirements in accordance with the applicable regulation ensures that NRC review and approval will be proposed for changes exceeding the stated regulatory threshold (e.g., an unreviewed safety question or a decrease in effectiveness). Elimination of reporting requirements that are recommended for relocation or removal from the TS can be proposed if they are not required by 10 CFR 50.72, 10 CFR 50.73 or other regulations.

3.1 Table of Contents

The proposed changes to the Table of Contents reflect the deletion of TS.3.1.E and TS.4.4.D and the bases associated with each specification, reformatting of Section 6, and excluding the bases citations.

The proposed changes to the Table of Contents are administrative only and reflect the proposed changes discussed in this safety evaluation (SE). The removal of the Bases Table of Contents reflects that the bases are controlled separately and are not part of the TS. This is consistent with the STS. Some minor pagination errors on page TS-viii were identified by the staff and corrected after telephone discussion with D. Vincent (NSP) on September 17, 1998. Therefore, the proposed changes to the Table of Contents are acceptable to the staff.

3.2 Maximum Coolant Activity Report

TS 3.1.D.4, Maximum Coolant Activity, currently requires annual reporting requirements in accordance with TS 6.7.A.1.c. The proposed change deletes this reference to annual reporting requirements, since the licensee proposes to delete TS 6.7.A.1.c. The deletion of TS 6.7.A.1.c is discussed in Section 3.17 below. The Maximum Coolant Activity Report is not required to be in the TS under the criteria of 10 CFR 50.36(c)(2)(ii).

The proposed change to TS 3.1.D.4 is administrative only and ensures consistency with the proposed changes to TS 6.7.A.1.c. This change is therefore acceptable to the staff.

3.3 Chemistry

The licensee proposes to delete TS 3.1.E, "Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration," in its entirety. TS 3.1.E provides limits on chemical concentrations in the reactor coolant system (RCS) to protect plant equipment and the pressure boundary. The RCS chemistry limits for maximum reactor coolant oxygen, chloride, and fluoride concentration shall be relocated to the Updated Safety Analysis Report (USAR) by the next USAR update, but no later than June 1, 1999. This relocation requirement has been included as a condition in Appendix B to the licenses.

The reactor coolant water chemistry limits on particular chemical properties of the primary coolant, and surveillance practices to monitor those properties, ensure that degradation of the reactor coolant pressure boundary and equipment is not exacerbated by poor chemistry conditions. However, degradation of the reactor coolant pressure boundary is a long-term process, and there are other more direct means to monitor and correct the degradation of the reactor pressure boundary and RCS equipment that are controlled by regulations and TS; for example, in-service inspection conducted in accordance with 10 CFR 50.55a, and primary coolant leakage limits. On this basis, the staff has concluded that the reactor coolant chemistry limits are not required to be in the TS to protect public health and safety and may be relocated to licensee procedures and included in the USAR. The TS is deleted, consistent with STS, since it does not meet the criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. The staff finds this change acceptable.

3.4 Minimum Frequencies for Sampling Tests

This proposed change deletes item 7, "RCS Chemistry," in Table TS.4.1-2B, "Minimum Frequencies for Sampling Tests," to ensure consistency with the change to TS 3.1.E., "Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration." The licensee proposed removal of the footnote associated with the * at the bottom of the page.

The staff determined that the footnote should be retained since it is applicable to another entry on the table. The proposed change to Table TS.4.1-2B to delete the entry for RCS Chemistry (CL, F, O2) is administrative only and ensures consistency with the deletion of TS.3.1.E as discussed in Section 3.3 of this SE. Table TS 4.1-2B, item 7, does not meet the criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. Therefore, the staff finds this deletion acceptable.

3.5 Residual Heat Removal (RHR) System

TS 4.4.D, "RHR System," currently requires portions of the RHR system external to the isolation valves at the containment be hydrostatically tested for leakage during each refueling shutdown, visually inspected for leakage, quantifying leakage to ensure that the maximum leakage at 350 psig does not exceed 2 gallons per hour, completing repairs if the leakage limit is exceeded, and shutting down and depressurizing the unit if repairs are not completed within 7 days. The licensee proposes to delete this TS and encompass the requirements in the program referenced in proposed TS 6.5.B, "Primary Coolant Sources Outside Containment."

The details of TS SR 4.4.D on RHR system leakage limits can be deleted because the limits are encompassed in the proposed Administrative Controls Section TS 6.5.B. This change is consistent with STS. TS 4.4.D does not meet the criteria of 10 CFR 50.36(c)(2)(ii) for inclusion in the TS. Accordingly, the TS change is acceptable.

3.6 Sampling of Diesel Fuel Oil

Currently, TS 4.6, "Periodic Testing of Emergency Power System," Section A, "Diesel Generators," item 1.c, requires that at least once a month, for each diesel generator, a sample of diesel fuel from the fuel storage tank is verified to be within the acceptable limits specified in Table 1 of American Society for Testing and Materials (ASTM) D975-77 when checked for viscosity, water, and sediment. The licensee proposes to delete this TS and relocate requirements similar to the STS to proposed TS Section 6.5.K, "Diesel Fuel Oil Testing Program." This relocation requirement has been included as a condition in Appendix B to the licenses. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with the limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment. Acceptability of new fuel oil shall be determined prior to addition to the safeguards storage tanks. Testing of diesel fuel oil stored in safeguards storage tanks shall be performed at least every 31 days. The provisions of TS 4.0 are applicable to the diesel fuel oil testing programs surveillance frequencies.

The proposed change is consistent with STS for the inclusion of a program for diesel fuel oil testing and is acceptable. Until fuel oil TS are proposed to meet the LCOs and SRs in accordance with the STS, the SRs in the current TS are proposed for deletion and incorporated as additional information in the proposed fuel oil testing program administrative controls.

3.7 Emergency Procedures

TS 5.1, "Design Features," currently states that TS 6.5.A.7 requires an emergency procedure that necessitates plant shutdown for flood water levels above +692 feet mean sea level (MSL) at the plant site. The emergency procedures will assure the proper erection of flood protection panels and assure an orderly shutdown of the plant and protection of safety-related facilities. This procedure must provide for progressive action levels to prevent the possibility of unsafe plant operation and must include requirements for periodic inspection of flood protection measures. Additionally, emergency procedures prepared in accordance with TS 6.5.A.7 must define actions required for earthquakes, including plant shutdown and inspection if an operational basis earthquake is measured at the site.

The licensee proposes to delete the description of emergency procedures for floods and earthquakes and the criteria for plant shutdown in response to these events since the proposed changes to TS 6.4, "Procedures," requires written procedures shall be established, implemented, and maintained covering applicable procedures in Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements," Revision 2, Appendix A, February 1978.

The details of Design Features TS 5.1 on the emergency procedures for floods and earthquakes can be deleted because the requirements are encompassed in the Administrative Controls Section 6.4 of the TS. This change results in comparable restrictions. The flood level that requires action to be taken to shut down the plant shall be included in the USAR by the next USAR update, but no later than June 1, 1999. Accordingly, the staff finds the changes to be acceptable.

3.8 Responsibilities

The current title of TS Section 6.1, "Organization," is being changed to "Responsibility." The first paragraph of the current TS Section 6.1 is revised to conform with STS with the following exceptions. The title "plant manager" is used instead of Plant Superintendent since it is the title currently in use at Prairie Island for the position with overall responsibility. The capitalization on the title "plant manager" has been changed to lower case to indicate that this is a generic title associated with the responsibilities rather than a plant-specific title. These changes are administrative and consistent with the licensee's organization. Therefore, they are acceptable to the staff.

The second paragraph of Section 6.1, added in conformance with STS, provides further description of the plant manager's responsibilities and is acceptable to the staff.

The third paragraph of Section 6.1, added in conformance with STS, provides requirements for the control room command function and is acceptable to the staff.

The content of the Administrative Controls TS 6.1 is editorially updated to reflect current position titles and enhanced with additional information on the plant manager's responsibilities and the control room command function. These TS changes are administrative, are consistent with the improved STS and the licensee's organization, and are acceptable.

3.9 Organization - Onsite and Offsite Organizations

Current TS 6.1, "Organization," that includes the following sections:

TS 6.1.A	plant manager function,
TS 6.1.B	Onsite and Offsite Organizations,
TS 6.1.C	Plant Staff,
TS 6.1.D	plant staff qualifications, and
TS 6.1.E	administrative procedures (shift coverage, overtime control),

is proposed to be relocated to TS 6.2, "Organization." TS 6.2 consists of TS 6.2.A, "Onsite and Offsite Organization," and TS 6.2.B, "Plant Staff," to conform with the organization of the STS. Section 6.2 conforms in content with STS with the following exceptions that are evaluated as follows.

When both units are collectively under consideration, "plant" had been substituted for "unit." This clarifies the TS and is acceptable.

Throughout the proposed changes, generic titles describing responsibilities are used. The intent of these changes is to allow the licensee to make specific title changes by providing updates to the USAR. Proposed TS 6.2.A.1 states, "the plant specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Updated Safety Analysis Report." The intent of this statement is to allow the licensee to make title changes and provide appropriate documentation in the USAR. This provision also applies to the corporate officer position identified in Specification 6.2.A.3. Incorporation of organizational titles in the USAR is acceptable to the staff, in that changes are adequately controlled by 10 CFR 50.54(a)(3).

The current TS 6.2.B.1 takes exception to 10 CFR 50.54 in that, when one unit is operating and the other unit is not operating, only two licensed reactor operators are required on site (in addition to two senior reactor operators.) Prairie Island proposes to delete the second sentence of TS 6.2.B.1 and thereby clarify conformance with the requirements of 10 CFR 50.54. This change is acceptable.

The content of proposed TS 6.2 is editorially updated to reflect current position titles and terminology. The requirements related to crew composition can be deleted from TS 6.2, because these requirements are adequately addressed in 10 CFR 50.54(k), (l), and (m). The regulations describe the minimum shift composition for operating modes, as well as for cold shutdown and refueling. These requirements are described in the administrative instructions implementing 10 CFR 50.54 and in the Emergency Plan. The staff concludes that the shift staffing requirements in 10 CFR 50.54, in conjunction with the organizational responsibilities and authorities retained in the administrative controls, provide adequate control over the plant staffing requirements. These TS changes are consistent with the STS and are acceptable.

The proposed TS 6.2.B.5 will take the place of current TS 6.1.E, which will be deleted. Prairie Island currently uses a staff overtime control program that was reviewed and approved by NRC Safety Evaluation under cover letter dated March 17, 1983 from D. Vassallo (NRC) to D. Musolf (NSP) and License Amendments 105/98. The wording of proposed TS 6.2.B.5 is consistent with one of the STS options and is acceptable.

3.10 Review and Audit

The licensee proposes that the review and audit functions associated with the Safety Audit Committee (SAC) and the Operations Committee (OC) specified in existing TS 6.2 be relocated from the TS to the Operational Quality Assurance Plan (OQAP) such that future changes could be made pursuant to 10 CFR 50.54(a). Section 13.4, "Operational Review", of NUREG-0800, the "Standard Review Plan" (SRP), provides the acceptance criteria used by the staff to evaluate TS provisions related to the plant staff review of operational activities performed by licensee organizational units fulfilling the review and audit function. This acceptance criteria is based on meeting the relevant requirements of 10 CFR 50.40(b) as it relates to the licensee being technically qualified to engage in licensed activities, and of Appendix B to 10 CFR Part 50 as it relates to the review and audit functions required by the licensee's QA program. Therefore, TS provisions associated with the review and audit function satisfy the criteria in both Section 50.36(c)(5), and Appendix B to 10 CFR Part 50. These provisions do not satisfy the criteria for inclusion in TS of 10 CFR 50.36 (c)(2)(ii) and can be relocated to the licensee's QA program description, consistent with NRC Administrative Letter 95-06. Additionally, the following considerations support relocating these items from the TS:

1. The licensee has proposed that the Prairie Island SAC function, membership, qualifications, meeting frequency, quorum, responsibilities, authority, and records provisions be relocated, verbatim, to Section 21.0, "Prairie Island SAC," of Revision 20 to the OQAP. Subsequent changes associated with SAC requirements will be controlled effectively under 10 CFR 50.54(a).

2. The licensee has proposed that the Prairie Island OC membership, meeting frequency, quorum, responsibilities, authority, records, and procedures provisions be relocated, verbatim, to Section 22.0, "Prairie Island OC," of Revision 20 to the OQAP. Subsequent changes associated with OC requirements will be controlled effectively under 10 CFR 50.54(a).

This approach is consistent with NRC Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995, that provides guidance for relocating TS administrative requirements. This approach would also result in an equivalent level of regulatory authority while providing for an acceptable change control process under the provisions of 10 CFR 50.54 (a)(3). On this basis, the staff has concluded that the review and audit functions identified above are not required to be included in the TS to protect public health and safety and may be relocated to the OQAP.

3.11 Special Review and Audit

The licensee proposes to relocate the provisions in the existing TS 6.3, "Special Inspections and Audits," to the Fire Protection Program. TS 6.3 requires an annual inspection and audit by qualified personnel and a triennial inspection and audit performed by a qualified fire protection consultant. The licensee will incorporate a 2-year limit on performance-based audit schedules in accordance with American National Standards Institute (ANSI) N-18.7, which is committed to in the licensee's OQAP, and retains the existing frequency for audits of the fire protection program on a fixed basis in accordance with Generic Letter (GL) 88-12, "Removal of Fire Protection Requirements from Technical Specifications." The relocation provides adequate controls in accordance with 10 CFR 50.54(a) and is acceptable to the staff.

3.12. Plant Staff Qualifications

Proposed TS 6.3, "Plant Staff Qualifications," is being added to more closely conform with STS. TS 6.3 includes current TS 6.1.d with one revision. The revision includes current plant practice that personnel performing the function of shift technical advisor (STA) shall maintain an active senior reactor operator license. Prairie Island will continue to commit to RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 1, September 1975 and ANSI N18.1-1971 that is also endorsed by the regulatory guide, as stated in the licensee's OQAP. This relocation results in identical restrictions.

The staff finds the change requiring the STA to have a senior reactor operator license acceptable since this is a desirable requirement for those performing the STA function. The staff finds the rewording included in the September 4, 1997, submittal acceptable since the wording now conforms with the most recent STS wording. Accordingly, the TS change is administrative and acceptable.

3.13 Safety Limit Violation

TS Section 6.4, "Safety Limit Violation," was previously relocated to TS 2.2 by Amendment Nos. 123/116, issued May 21, 1996, and is not addressed by these amendments.

3.14 Procedures

The licensee proposes to relocate portions of current TS 6.5 to TS 6.4, "Plant Operating Procedures," to conform to Section 5.4, "Procedures" of NUREG-1431, Revision 1. The proposed TS 6.4 includes the following sections:

6.4	Procedures,
6.4.A	Procedures required by RG 1.33, Rev 2, Appendix A,
6.4.B	EOPs [emergency operating procedures] per NUREG-0737 and NUREG-0737 Supplement 1,
6.4.C	Quality control for effluent and environmental monitoring,
6.4.D	Fire protection program implementation, and
6.4.E	All programs specified in TS 6.5.

The information in TS 6.4.A includes the following procedures that are acceptable to the staff in accordance with RG 1.33, Revision 2, Appendix A as follows:

Current TS 6.5.A.1 is addressed by RG 1.33, Rev. 2, Appendix A, Section 3, Procedures for Startup, Operation, and Shutdown of Safety-Related PWR [Pressurized Water Reactor] System.

Current TS 6.5.A.2 is addressed by RG 1.33, Appendix A, Section 2, General Plant Operating Procedures, Items k and l and Section 6, Procedures for Combating Emergencies and Other Significant Events, Item x.

Current TS 6.5.A.3 is addressed by RG 1.33, Rev. 2, Appendix A, Section 2, General Plant Operating Procedures, Item c; Section 5, Procedures for Abnormal, Offnormal, or

Alarm Conditions; and Section 6, Procedures for Combating Emergencies and Other Significant Events.

Current TS 6.5.A.4 is addressed by RG 1.33, Rev. 2, Appendix A, Section 8, Procedures for Control of Measuring and Test Equipment and for Surveillance Tests, Procedures, and Calibrations, Item B.

Current TS 6.5.A-6 is addressed by RG 1.33, Rev. 2, Appendix A, Section 6, Procedures for Combating Emergencies and Other Significant Events, Item w. The requirement to shut down the plant if the flood level reaches 692 feet above MSL will be relocated to the USAR.

Current TS 6.5.B is addressed by RG 1.33, Rev. 2, Appendix A, Section 7, Procedures for Control of Radioactivity, Item 7, and the requirements of 10 CFR Part 20.

Current TS 6.5.C, "Maintenance and Test," is generally included in RG 1.33, but the specific list will be relocated to the OQAP.

Current TS 6.5.A.5 and TS 6.5.B.3 will be relocated to the Emergency Plan, since the requirement for written procedures that implement the Emergency Plan are specified in 10 CFR 50.54(q) and (t), and 10 CFR Part 50, Appendix E, Section V; the airborne iodine measurement program is embodied in the Emergency Plan; and changes to the Emergency Plan are controlled by 10 CFR 50.54(q) and 50.4. This relocation is consistent with STS and is acceptable to the staff.

Current TS 6.5.A-8 and TS 6.5.D were relocated by TS Amendment Nos. 122/115, issued January 24, 1996.

By letter dated April 4, 1996, NSP submitted a supplement further clarifying its previous submittals. The revisions included TS 6.4.B concerning the requirements for EOPs that now includes the base document NUREG-0737 in addition to NUREG-0737, Supplement 1. The staff finds this acceptable since the wording now conforms to STS.

The licensee proposes to revise current TS 6.5, "Plant Operating Procedures," to conform to Section 5.4, "Procedures" of NUREG-1431, Revision 1. The revised section, TS 6.4, "Procedures," would include only procedural requirements for activities addressed in NUREG-1431, and all other procedure-related provisions currently in TS 6.5 would be relocated, verbatim, to the OQAP, Section 7.3, "Procedures." The relocation provides adequate controls in accordance with 10 CFR 50.54(a), and is acceptable to the staff.

TS 6.5 requirements related to the plant Emergency Plan would continue to be controlled through 10 CFR 50.54(q) and 50.54(t) in accordance with GL 93-07, "Modification of the Technical Specification Administrative Control Requirements for Emergency and Security Plans." Additionally, the licensee proposed that TS 6.5.C, "Temporary Changes to Procedures," be relocated, verbatim, to the OQAP, Section 8.4.4, "Prairie Island Procedure Control." Therefore, subsequent changes to these provisions would be controlled pursuant to 10 CFR 50.54(a).

Based on the above, the staff concludes that the proposed TS 6.4 would conform to the format and content of NUREG-1431, Revision 1, and that procedural requirements currently in TS 6.5, but not addressed in STS, could be relocated as previously described to either the OQAP or the Emergency Plan. Therefore the staff finds this change acceptable.

3.15 Programs and Manuals

Proposed TS Section 6.5, Programs and Manuals, includes the following:

6.5.A This section consists of current TS 6.5.E, "Offsite Dose Calculation Manual."

6.5.B Current TS 6.5.B.2, "Primary Coolant Sources Outside Containment," is relocated to this section. This section also encompasses current TS 4.4.D, "Residual Heat Removal Tests."

6.5.C Current TS 6.5.B.3 and TS 6.5.B.4, "Post Accident Sampling," are relocated to this section.

6.5.D Current TS 6.5.H, "Radioactive Effluent Controls Program," is relocated to this section.

6.5.E Current TS 6.6.B.8, "Component Cyclic or Transient Limit," is relocated to this section.

6.5.F Current TS 6.5.F, "Security," is being deleted since the program is covered by 10 CFR 50.54(p)(1) and 10 CFR 73.55. The licensee proposes reserving this section for future use.

6.5.G Current TS 6.5.G, "Temporary Changes to Procedures," is being deleted from the TS and included in the OQAP. The licensee proposes reserving this section for future use. The relocation provides adequate controls in accordance with 10 CFR 50.54(a), and is acceptable to the staff.

6.5.H Current TS 6.5.H, "Radioactive Effluent Controls Program," is being relocated to 6.5.D. The licensee proposes to reserve this section for the Steam Generator Tube Surveillance Program that currently is included in TS 4.12.

6.5.I The licensee proposes to reserve this section for the ventilation filter testing program that is currently covered in TS 4.4, 4.14, and TS 4.15.

6.5.J This section will consist of the current TS 6.5.I, "Explosive Gas and Storage Tank Radioactivity Monitoring Program."

6.5.K The Diesel Fuel Oil Testing Program will be relocated from current TS 4.6.A.1.c

6.5.L TS Bases Control Program is new.

6.5.M Containment Leakage Rate Testing Program is relocated from current TS.6.5.J

Proposed TS 6.5.A, TS 6.5.B, TS 6.5.C, TS 6.5.D, TS 6.5.E, TS 6.5.J, TS 6.6.K, and TS 6.5.M are relocations of existing requirements and are acceptable to the staff. The licensee proposes a correction to the quantity of radioactivity contained in each gas storage tank from 78,000 to 78,800 curies. This change corrects a typographical error introduced in TS 6.5.I.2 during Amendment Nos. 122/115, issued January 24, 1996, and is acceptable to the staff.

The licensee proposed to relocate the requirements for both the review and approval process in TS 6.8.2, and the temporary change process for procedures in TS 6.8.3, to the OQAP. The relocation provides adequate controls in accordance with 10 CFR 50.54(a), and is acceptable to the staff. The revised TS will include a specific requirement that written procedures be established, implemented, and maintained, and a requirement for procedure control is mandated by 10 CFR Part 50, Appendix B, Criterion II and Criterion V. ANSI N18.7-1976, an NRC staff-endorsed document used in the development of many licensee QA plans, also contains specific provisions related to procedures. The licensee has incorporated certain provisions of ANSI N18.7-1976 and ANSI N45.2-1971 in the OQAP, as stated in OQAP Section 1.3, as a means to comply with 10 CFR Part 50, Appendix B. ANSI N18.7-1976, Section 5.2.2 discusses procedure adherence. This section clearly states that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. ANSI N18.7-1976 also discusses temporary changes to procedures and requires review and approval of procedures to be defined. ANSI N18.7-1976, Section 5.2.15, describes the review, approval, and control of procedures. This section states that the licensee's OQAP is to provide measures to control and coordinate the approval and issuance of documents, including changes thereto, that prescribe all activities affecting quality. The section further states that each procedure shall be reviewed and approved prior to initial use. The required reviews are also described. ANSI N45.2-1971, Section 6, also prescribes procedural controls for activities affecting quality.

The provisions in the OQAP implement 10 CFR Part 50, Appendix B, Criteria V and VI, pertaining to the control of documents such as instructions, procedures, and drawings, including changes thereto. The procedure review and approval functions currently in TS define an administrative framework to ensure that documents are reviewed for adequacy and approved for release by authorized personnel. The required control of these processes in 10 CFR Part 50, Appendix B, Criteria V and VI and the revised OQAP is considered to be redundant and functionally equivalent to the provisions currently in TS. The staff has determined that the procedure review and approval functions are adequately addressed by 10 CFR Part 50, Appendix B, and the related OQAP changes. Based upon the relocation of the procedure review provisions to the OQAP, it is not necessary to include redundant or additional requirements in the TS administrative controls.

The licensee will continue to implement an OQAP in accordance with the requirements of 10 CFR Part 50, Appendix B, Criteria V and VI, that requires appropriate controls for the review and approval of procedure changes. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing them from the TS is acceptable. Future changes to the review and approval process for procedure changes can be adequately controlled under 10 CFR 50.54(a).

The licensee proposes to add TS 6.5.L, "Technical Specifications Bases Control Program," that generally follows the guidance of STS and is acceptable to the staff. In TS 6.5.L.1, the licensee proposes to correct a typographical error. The phrase, "Bases or the [TS]..." should read, "Bases of the [TS]..." The correction is editorial and is acceptable to the staff.

3.16 Plant Operating Records

The licensee proposes to relocate, verbatim, the record retention requirements in TS 6.6, "Plant Operating Records," to OQAP Section 19.12, "Prairie Island Operating Records," in accordance with Administrative Letter 95-06. Once relocated to the OQAP, these record retention requirements are adequately addressed by 10 CFR 50.54(a). In addition to specific record retention commitments in the OQAP, the licensee relies upon its OQAP commitments to ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," and ANSI N45.2.9-1974, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants," (as endorsed by RG 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," Revision 2) to satisfy the regulatory requirements of 10 CFR Part 50, Appendix B, Criterion XVII, "Quality Assurance Records."

Since current TS 6.6 is being relocated, the licensee proposes to delete the specification in its entirety based on duplication of OQAP commitments and the requirements of 10 CFR Part 20, Subpart L, and 10 CFR 50.71. This relocation is consistent with STS.

The licensee proposed that the requirements for record retention in TS 6.10 be relocated because they are adequately addressed by the OQAP. The provisions in the OQAP implement 10 CFR Part 50, Appendix B, Criterion XVII pertaining to the maintenance of records related to activities affecting quality. The required controls related to record retention specified in various regulations and the provision incorporated into the OQAP are considered to be redundant to the requirements currently in TS. The staff has determined that record retention requirements are adequately addressed by 10 CFR Part 50, Appendix B, Criterion XVII and the related OQAP commitments. Based upon the relocation of the record retention provisions to the OQAP, it is not necessary to include redundant or additional requirements in the TS administrative controls. The relocation provides adequate controls in accordance with 10 CFR 50.54(a) and is acceptable to the staff.

The staff concludes that the regulatory requirements under 10 CFR Part 50, Appendix B, provide sufficient control of the plant records, and sufficient regulatory controls exist for future changes to the program pursuant to 10 CFR 50.54(a). In addition, other regulations such as 10 CFR Part 20, Subpart L, and 10 CFR 50.71 require the retention of certain records related to operation of the nuclear plant. The staff concludes that these regulatory requirements provide sufficient control of these recordkeeping provisions and removing them from the TS is acceptable.

3.17 Reporting Requirements

Proposed Section 6.6, Reporting Requirements, will include current TS Section 6.7 as detailed in the following evaluation. The evaluation discusses those Section 6.7 requirements that are relocated, deleted, and not provided in accordance with STS.

The STS Administrative Controls specifies Section 5.6, "Reporting Requirements." The licensee proposes the following sections under TS 6.6, "Reporting Requirements:"

A. Occupational Exposure Report,
B. Annual Radiological Environmental Monitoring Report,
C. Radioactive Effluent Report,
D. Monthly Operating Report,
E. Core Operating Limits Report (COLR),
F. Pressure and Temperature Limit Report.

The licensee proposes to relocate current TS 6.7.A.1.a, "Occupational Exposure Report," and its associated footnote and to revise the section to correspond to STS and include the section as proposed TS 6.6.A.

The licensee proposes to relocate current TS 6.7.A.1.b, "Report of Safety and Relief Valve Failures and Challenges," to the proposed TS 6.6.D, "Monthly Operating Report."

The licensee proposes to relocate current TS 6.7.C.1, "Annual Radiological Environmental Monitoring Report," to proposed TS 6.6.B. The proposed TS 6.6.B includes a clarification to the wording of the distance from one reactor to the sample location. The proposed wording is from "the site" to the sampling locations. As explained in the submittal, the center of the reactor site which is less than 200 feet from the centerline of either reactor will be used. The licensee also proposes to eliminate the word "all" from "a map of all sampling locations," since the wording is superfluous.

Current TS 6.7.A.4, "Radioactive Effluent Report," is proposed to be relocated to TS 6.6.C.

The licensee proposes to relocate current TS 6.7.D, "Special Reports," to proposed TS 6.6.D, "Monthly Operating Report." An incorrect reference regarding whom to submit the report is corrected.

Current TS 6.7.A.6, "Core Operating Limits Report," is proposed to be relocated to TS 6.6.E.

The relocation of current TS 6.7.A.1.a, 6.7.A.1.b, 6.7.C.1, 6.7.A.4, 6.7.D, and 6.7.A.6 to proposed TS 6.6.A, 6.6.B, 6.6.C, 6.6.D, and 6.6.E is administrative and the corrections made are acceptable.

The licensee proposes to delete the following reports included in current TS 6.7:

Current TS 6.7.A.1.c, "Primary Coolant Iodine Spike Report," is deleted because the report is unnecessary. The original purpose of the report was to provide to the NRC baseline data on core iodine levels in the commercial nuclear industry for use in postulating accident iodine releases. With the maturing of the industry, this data is well established and this report is unnecessary. In addition, serious degradation of a fission product barrier is required to be reported by 10 CFR 50.73. The deletion of this report is administrative and does not affect plant operation.

Current TS 6.7.A.2, Startup Reports, is deleted because the report is unnecessary. Startup Reports, as required by TS 6.7.A.2, are required when an operating license is received, plant power level is increased by license amendment, fuel of a different design or manufacturer is installed, or modifications are performed which significantly alter the nuclear, thermal, or hydraulic performance of the plant. All of these plant changes are accompanied by specific NRC authorization, and requirement for a report would appropriately be addressed in the concomitant TS or license condition. The Startup Report is not required to ensure safe plant operation. The approved 10 CFR Part 50, Appendix B, QA Plan, and FSAR startup testing program provide assurance that the affected activities are adequately performed and that appropriate corrective actions, if required, are taken.

Current TS 6.7.A.5, Annual Summaries of Meteorological Data, is being deleted, consistent with STS.

Current TS 6.7.D, Special Reports, is being deleted, consistent with STS.

Over the last several years, there were several initiatives to reduce unnecessary administrative burdens associated with reporting requirements, while retaining an appropriate level of publicly accessible information flow. The staff has concluded that the reports are unnecessary because the information is duplicated in other required reports, such as reports required by the Offsite Dose Calculation Manual, Radioactive Effluent Controls Program, and Radiological Environment Monitoring Program, or the reports are not required for the safe operation of the plant. In addition, the notification requirements in 10 CFR 50.72 and 50.73 for plant conditions that may be safety significant or warrant emergency response address these matters.

Accordingly, the staff has concluded that the Primary Coolant Iodine Spike Report, Startup Reports, Annual Summaries of Meteorological Data, and Special Reports are unnecessary because the information is duplicated in other required reports or the information is not required for the safe operation of the plant.

TS 6.7.C.3, Other Environmental Reports, is not required for safe operation of the plant and is deleted consistent with STS.

The licensee proposes that the requirement in TS 6.7.B, that the Commission be notified of all reportable events, be deleted from the TS on the basis that this requirement is adequately addressed in the regulations. Requirements are provided in 10 CFR 50.73(a)(2) for the licensee to submit a Licensee Event Report (LER) for all reportable events specified in 10 CFR 50.73. The staff concludes that these reporting requirements are sufficient and removing the duplicative reporting requirements from the TS is acceptable.

Reports included by STS, but not included in this submittal are:

Proposed TS 6.6 does not include an EDG [emergency diesel generator] failure report since this requirement is not included in current TS.

Proposed TS 6.6 does not include provision for reporting of Post-Accident Monitoring Instrumentation failures because reporting requirements are included in current TS 3.15.

Proposed TS 6.6 does not include a steam generator tube inspection report requirement since current TS 4.12, "Steam Generator Tube Surveillance," Item E, "Reports," is being retained.

Since there is no proposed change to the current TS, no discussion is required. However, to meet the format of the STS, proposed changes to address these reporting requirements are expected when revisions to TS Sections 3/4 are submitted.

3.18 High Radiation Area

Current TS 6.5.B.1 is relocated to proposed TS 6.7, High Radiation Areas, consistent with the format of the STS administrative controls section. The licensee proposes to add to TS 6.7.A "or equal to" to the definition of high radiation area as follows: "each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr but less than or equal to 1000 mrem/hr...." Similarly, the licensee proposes to delete "or equal to" prior to 1000 mr/hr from TS 6.7.B so that it reads as follows: "areas with radiation levels greater than 1000 mrem/hr...."

The proposed changes are editorial and reflect the wording in the STS. The staff finds the changes acceptable.

3.19 Chemistry Bases

The licensee proposes to delete the current TS Bases, 3.1.E, "Reactor Coolant System Maximum Coolant Oxygen, Chloride, and Fluoride Concentration," since LCO 3.1.E is being deleted.

The proposed deletion of TS Bases, 3.1.E is administrative only and is consistent with the deletion of TS.3.1.E as discussed in Section 3.3. The staff finds the proposed change acceptable.

3.20 RHR System Bases

The licensee proposes to delete the Bases for current TS 4.4.D, "Residual Heat Removal System," since TS 4.4.D is being deleted. The provisions of TS 4.4.D are to be relocated to proposed TS 6.5.B.

As discussed in Section 3.5, the staff found the proposed change to TS 4.4.D to be acceptable. The deletion of TS Bases 4.4.D provides consistency with the proposed changes. Accordingly, the TS change is administrative and acceptable to the staff.

3.21 Summary

The staff has evaluated the relocation of some of those administrative controls to the OQAP. Based on this evaluation, the staff has concluded that (1) the proposed Prairie Island TS 6.4, "Procedures," conforms to the format and content specified in NUREG-1431, Revision 1, and to the requirements of 10 CFR 50.36(c)(5); (2) the proposed relocation of QA-related administrative control provisions (Section 6.2, "Review and Audit," Subsection C, "Temporary Changes to Procedures," of Section 6.5, "Plant Operating Procedures," and Section 6.6, "Plant

Operating Records") from the TS to the OQAP satisfies Administrative Letter 95-06 provisions and 10 CFR 50.36 requirements and, once relocated to the OQAP and controlled pursuant to 10 CFR 50.54(a), constitute the bases for the licensee's continued compliance with the requirements of Appendix B to 10 CFR Part 50; and (3) Revision 20 to the OQAP, dated July 14, 1997, continues to comply with the criteria of Appendix B to 10 CFR Part 50 in accordance with NUREG-0800 (SRP Sections 13.4 and 17.2).

In conclusion, the existing TS requirements relating to administrative controls that have been deleted or relocated are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act and are governed by other regulations such as 10 CFR 50.4, 50.47, 50.48, 50.54, 50.72, 50.73, Part 50 Appendix A, Part 50 Appendix B, Part 50 Appendix E, Part 20, Part 55, or 73.55. Thus, the relocated provisions do not meet the intent of the four criteria described in the Commission's Final Policy Statement and included in 10 CFR 50.36(c)(2). In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59, 50.54(a), (k), (l), (m), (p), (q), and (t) and 73.55 to control future changes to the relocated provisions.

Accordingly, the staff has concluded that these requirements may be relocated from the TS to the above specified documents. Finally, the staff concludes that the administrative controls requirements remaining in the TS satisfy the license content specified in 10 CFR 50.36(c)(5).

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 requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration and there has been no public comment on such findings (61 *FR* 28618). The amendments also change recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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 Contributors: T. Tjader
J. Peralta
C. Goodman

Date: December 7, 1998

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

Carl F. Lyon, 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. 141 to DPR-42
2. Amendment No. 132 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

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