DCR 016

June 25, 1985

Docket Nos. 50-282
and 50-306

DISTRIBUTION:

Docket File	PMKreutzer-3
NRC PDR	DCDiIanni
L PDR	Gray File +4
SECY	ACRS-10
ORB#3 Rdg	WJones
HThompson	TBarnhart-8
OELD	RDiggs
CMiles	LJHarmon
EJordan	DBrinkman
JPartlow	

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment Nos. 73 and 66 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, in response to your application dated July 11, 1984 as supplemented by letter dated April 26, 1985. By letter dated January 21, 1985, you withdrew your request for revision of the peaking factor limits which were included in the July 11, 1984 application.

The amendments:

1. Make administrative changes dealing with reporting requirements as defined in GL 83-43 related to 10 CFR 50.73 (TS 1-1 and other sections of TS);

2. Revise table of contents for the technical specifications (TS i thru viii);

3. Revise the 2000 ppm Boron concentration requirements to agree with standard technical specifications requirements during refueling operation (TS 1.P.3, TS 3.6.A.4 and TS 3.8.A.4);

4. Make administrative changes deleting the reference to TS section 6.7 for the reporting requirements related to the radioactive source leakage test (TS 4.11.D);

5. Add the restriction related to having an SRO in the Control Room at all times above cold shutdown as per 10 CFR 50.54(m)(2) (TS 6.1-1 table);

6. Delete the table listing the safety related snubbers as permitted by GL 84-13 (TS 3.12-1 table and TS 4.13-1);

7. Clarify the volume and concentration of Sodium Hydroxide requirement in the containment spray additive tanks (TS 3.3.B.1.c);

8. Clarify the requirement for measuring the plant discharge flow rate dealing with radioactive effluent requirements (TS 3.9-1);

9. Delete unnecessary testing related to radioactive effluent monitor instrumentation (TS 3.9-2 table, TS 4.17-1 table);

10. Clarify TS wording related to radiation environmental monitoring program sampling (TS 4.10-1 table); and

11. Change the title of the Director of Nuclear Generation to Vice President Nuclear Generation (TS 6.2 and TS 6.1-1 Figure).

As requested in your letter of January 21, 1985, we have not considered your proposed change related to the $F^N_{\Delta H}$ and F_Q limits.

Your proposed technical specification change associated with the reactor shutdown margin during refueling operations was modified in order that it would be consistent with the PWR Standard Technical Specifications. The modification was discussed with and agreed to by your staff.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next monthly <u>Federal Register</u> notice.

Sincerely,

/S/

Dominic C. Di Ianni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 73 to DPR-42
2. Amendment No. 66 to DPR-60
3. Safety Evaluation

cc w/enclosures:
See next page

RAB
WF Congel
5/23/85

WHM 5/23/85

CPB
R Lobel
5/24/85

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PMKreutzer	DCDi Ianni	JRMiller		GCLainas
5/[illegible]/85	5/21/85	5/21/85	6/[illegible]/85	6/[illegible]/85

Gerald Charnoff, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, NW
Washington, DC 20036

Executive Director
Minnesota Pollution Control Agency
1935 W. County Road, B2
Roseville, Minnesota 55113

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

Jocelyn F. Olson, Esq.
Special Assistant Attorney General
Minnesota Pollution Control Agency
1935 W. County Road, B2
Roseville, Minnesota 55113

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Route #2, Box 500A
Welch, Minnesota 55089

Regional Administrator
USNRC, Region III
Office of Executive Director for
Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
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 July 11, 1984, as supplemented April 26, 1985, 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 (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 73 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Edward J. Butcher

Edward J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 25, 1985

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
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 July 11, 1984, as supplemented April 26, 1985, 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 (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 66 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Edward J. Butcher

Edward J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: June 25, 1985

ATTACHMENT TO LICENSE AMENDMENTS NOS. 73 AND 66

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

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of changes.

Remove	Insert
i thru iv	i thru x
TS.1-1	TS.1-1
TS.1-6	TS.1-6
TS.3.1-12	TS.3.1-12
TS.3.3-3	TS.3.3-3
TS.3.6-2	TS.3.6-2
TS.3.6-5	TS.3.6-5
TS.3.8-1	TS.3.8-1
TS.3.8-3	TS.3.8-3
TS.3.8-4	TS.3.8-4
TS.3.9-1	TS.3.9-1
TS.3.9-3	TS.3.9-3
Table TS.3.9-1 (page 1 of 2)	Table TS.3.9-1 (page 1 of 2)
Table TS.3.9-2 (page 1 of 2)	Table TS.3.9-2 (page 1 of 2)
TS.3.12-1	TS.3.12-1
Table TS.3.12-1 (8 pages)	---
TS.3.14-1	TS.3.14-1
TS.3.14-2	TS.3.14-2
TS.3.14-3	TS.3.14-3
TS.3.14-4	TS.3.14-4
TS.3.15-1	TS.3.15-1
Table TS.4.10-1 (page 3 of 4)	Table TS.4.10-1 (page 3 of 4)

Remove	Insert
TS.4.11-2	TS.4.11-2
TS.4.12-5	TS.4.12-5
TS.4.13-1	TS.4.13-1
TS.4.13-2	TS.4-13-2
TS.4.13-3	TS.4.13-3
Table TS.4.17-1 (page 1 of 2)	Table TS.4.17-1 (page 1 of 2)
Table TS.4.17-2 (page 1 of 2)	Table TS.4.17-2 (page 1 of 2)
Table TS.6.1-1	Table TS.6.1-1
Figure TS.6.1-1	Figure TS.6.1-1
TS.6.2-1	TS.6.2-1
TS.6.2-3	TS.6.2-3
TS.6.2-5	TS.6.2-5
TS.6.2-6	TS.6.2-6
TS.6.7.3a	----
TS.6.7-4	TS.6.7-4
TS.6.7-5	TS.6.7-5
TS.6.7-6	----
TS.6.7-7	----
TS.6.7-8	----

TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TECHNICAL SPECIFICATIONS

LIST OF TABLES

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

1.0 DEFINITIONS

The succeeding infrequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Reportable Event

A Reportable Event shall be any plant occurrence or event which must be reported, per 10 CFR 50.73, requiring written reports to the Commission.

B. Deleted

3. Refueling Shutdown

A reactor is in the refueling shutdown condition when a refueling operation is scheduled, K_{eff} is equal to or less than 0.95, and the reactor coolant average temperature is less than 140°F.

Q. Thermal Power

Thermal power of a unit is the total heat transferred from the reactor core to the coolant.

R. Physics Tests

Physics tests are those conducted to measure fundamental characteristics of the core and related instrumentation. Physics tests are conducted such that the core power is sufficiently reduced to allow for the perturbation due to the test and therefore avoid exceeding power distribution limits in Specification 3.10.B.

Low power physics tests are run at reactor powers less than 5% of rated power.

S. Startup Operation

The process of heating up a reactor above 200°F, making it critical, and bringing it up to power operation.

T. Fire Suppression Water System

The fire suppression water system consists of: Water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

U. Minimum Pressurization Temperature (MPT)

Reactor coolant system temperature below which reactor coolant system pressure is limited by Figures TS.3.1-1 and TS.3.1-2, Reactor Coolant System Heatup and Cooldown Limitations.

4. If a reactor is at or above cold shutdown:

   (a) With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\overline{E}$ 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. A special report shall be submitted to the Commission within 30 days. This report shall contain the results of the specific activity analyses together with the following information:

      1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

      2. Fuel burnup by core region,

      3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

      4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

      5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

<u>Basis</u>

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.3.1-5, 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.3.1-5 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by

c. The spray additive tank contains not less than 2590 gallons of solution with a sodium hydroxide concentration of 9% to 11% by weight inclusive.

d. Manual valves in the above systems that could (if improperly positioned) reduce spray flow below that assumed for accident analysis, shall be blocked and tagged in the proper position. During power operation, changes in valve position will be under direct administrative control.

e. Automatic valves, interlocks, ducts, dampers, controls and piping associated with the above components and required for accident conditions are operable.

f. The following motor-operated valve conditions shall exist:

(1) The Unit 1 operation, containment spray system motor-operated valves MV32096 and MV32097 shall be closed and shall have the motor control center supply breakers open.

(2) For Unit 2 operation, containment spray system motor-operated valves MV32108 and MV32109 shall be closed and shall have the motor control center supply breakers open.

2. During startup operation or power operation, any one of the following conditions of inoperability may exist for each unit provided startup operation is discontinued until operability is restored. The reactor shall be placed in the hot shutdown condition if during power operation operability is not restored within the time specified. The reactor shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.

a. One fan cooler unit or one duct for a fan cooler unit may be out of service for a period not to exceed 48 hours. Prior to initiating repairs and once every 24 hours thereafter, both containment spray pumps and the remaining three fan cooler units shall be demonstrated to be operable.

b. One containment spray pump may be out of service for a period not to exceed 48 hours. The remaining containment spray pump and the four fan units shall be demonstrated to be operable before initiating repairs and once every 24 hours thereafter.

4. Positive reactivity changes shall not be made by boron dilution when containment system integrity is not intact unless the more restrictive of the following reactivity conditions is met: K_{eff} is $\leq$ 0.95 or the boron concentration $\geq$ 2000 ppm.

5. The vacuum breaker system shall be considered operable for containment system integrity when both valves in each of two vacuum breakers, including actuating and power circuits, are operable or when one vacuum breaker is daily demonstrated as operable and the other has been inoperable for no more than 7 days under conditions for which containment integrity is required.

6. Automatic containment isolation valves listed in Table TS.4.4-1 shall be considered operable for containment system integrity when all automatic isolation valves, including actuation circuits, for each penetration are operable or the inoperable valve is deactivated in the closed position, or at least one valve in each penetration having an inoperable valve is locked closed.

7. a. The 36-inch containment purge system double gasketed blind flanges shall be installed whenever the reactor is above cold shutdown.

   b. The 18-inch containment inservice purge system double gasketed blind flanges shall be installed whenever the reactor is above cold shutdown except as noted below.

   c. The inservice purge system may be operated above cold shutdown when required for safe plant operation if the following conditions are met:

      1. The debris screens are installed on the supply and exhaust ducts in containment.

      2. Both valves shall satisfactorily pass a local leak rate test prior to use.

      3. The two automatic primary containment isolation valves and the automatic shield building ventilation damper in each duct that penetrates containment shall be operable, including instruments and controls associated with them.

      4. The blind flanges (i.e., 42B (53 in Unit 2) and 43A (52 in Unit 2) shall be reinstalled and satisfactorily pass a local leak rate test, each time after the inservice purge system is used.

8. During maintenance, construction and testing activities, containment integrity is considered intact if the auxiliary building special vent zone boundary is opened intermittently, provided such openings are under direct administrative control and can be reduced to less than 10 square feet within 6 minutes following an accident.

This specification also prevents positive insertion of reactivity whenever containment integrity is not maintained if such addition would violate the respective shutdown margins. The boron concentration must be maintained at $\geq$ 2000 ppm and K_{eff} must be $\leq$ 0.95 if the containment system is to be disabled with the vessel open.

The 2 psig limit on internal pressure provides adequate margin between the maximum internal pressure of 46 psig and the peak accident pressure resulting from the postulated Design Basis Accident.(2)

The containment vessel is designed for 0.8 psi internal vacuum, the occurrence of which will be prevented by redundant vacuum breaker systems.

The containment has a nil ductility transition temperature of $0^{\circ}F$. Specifying a minimum temperature of $30^{\circ}F$ will provide adequate margin above NDTT during power operation when containment is required.

The conservative calculation of off-site doses for the loss of coolant accident(3)(6) is based on an initial shield building annulus air temperature of $60^{\circ}F$ and an initial containment vessel air temperature of $104^{\circ}F$. The calculated period following LOCA for which the shield building annulus pressure is positive, and the calculated off-site doses are sensitive to this initial air temperature difference. The specified $44^{\circ}F$ temperature difference is consistent with the LOCA accident analysis(6).

The initial testing of inleakage into the shield building and the auxiliary building special ventilation zone (ABSVZ) has resulted in greater specified inleakage (Figure TS 4.4-1, change No. 1) and the necessity to deenergize the turbine building exhaust fans in order to achieve a negative pressure in the ABSVZ (TS 3.6.A.10). The staff's conservative calculation of doses for these conditions indicated that changing allowable containment leak rate from 0.5% to 0.25%/day would offset the increased leakage.(5)

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The Charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these

3.8 REFUELING AND FUEL HANDLING

Applicability

Applies to operating limitations during fuel-handling and refueling operations.

Objectives

To ensure that no incident could occur during fuel handling and refueling operations that would affect public health and safety.

Specification

A. During refueling operations the following conditions shall be satisfied:

1. The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, at least one isolation valve shall be operable or locked closed in each line which penetrates the containment and provides a direct path from containment atmosphere to the outside.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool areas shall be monitored continuously.

3. The core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment, which are in service whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

4. During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration of the reactor coolant system and the refueling canal shall be sufficient to ensure that the more restrictive of the following reactivity conditions is met: $K_{eff} \leq 0.95$ or the boron concentration $\geq$ 2000 ppm. The required boron concentration shall be verified by chemical analysis daily.

5. During movement of fuel assemblies or control rods out of the reactor vessel, at least 23 feet of water shall be maintained above the reactor vessel flange. The required water level shall be verified prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.

6. At least one residual heat removal pump shall be operable and running. The pump may be shut down for up to one hour to facilitate movement of fuel or core components.

7. If the water level above the top of the reactor vessel flange is less than 20 feet, except for control rod unlatching/latching operations or upper internals removal/replacement, both residual heat removal loops shall be operable.

8. If Specification 3.8.A.6 or 3.8.A.7 cannot be satisfied, all fuel handling operations in containment shall be suspended, the containment integrity requirements of Specification 3.8.A.1 shall be satisfied, and no reduction in reactor coolant boron concentration shall be made.

Basis

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

Under rodded and unrodded conditions, the Keff of the reactor must be $\leq$ 0.95 and the boron concentration must be $\geq$ 2000 ppm as indicated in A.4. Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. A.9 above allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

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

The spent fuel assemblies will be loaded into the spent fuel cask for shipment to a reprocessing plant after sufficient decay of fission products. In loading the cask into a carrier, there is a potential drop of 66 feet.(4) The cask will not be loaded onto the carrier for shipment prior to a 3-month storage period. At this time, the radioactivity has decayed so that a release of fission products from all fuel assemblies in the cask would result in off-site doses less than 10 CFR Part 100. It is assumed, for this dose analysis, that 12 assemblies rupture after storage for 90 days. Other assumptions are the same as those used in the dropped fuel assembly accident in the SER, Section 15. The resultant doses at the site boundary are 94 Rems to the thyroid and 1 Rem whole body.

The Spent Fuel Pool Special Ventilation System(3) is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation System is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

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

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

References

(1) FSAR Section 9.5.2
(2) FSAR Section 14.2.1
(3) FSAR Section 9.6
(4) FSAR Page 9.5-20a

3.9 RADIOACTIVE EFFLUENTS

Applicability

Applies at all times to the liquid and gaseous radioactive effluents from the plant and the solidification and packaging for shipment of solid radioactive waste.

Objective

To implement the requirements of 10CFR20, 10CFR71, 10CFR50 Section 50.36a, Appendix A and Appendix I to 10CFR50, 40CFR141, and 40CFR190 pertaining to radioactive effluents.

Specifications

A. Liquid Effluents

1. Concentration

   a. The concentration of liquid radioactive material released at any time from the site (Figure 3.9-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} μci/ml total activity.

   b. When the concentration of radioactive material in liquid released from the site exceeds the limits in (a) above, immediately restore the concentration within acceptable limits.

2. Dose

   a. The dose or dose commitment to an individual from radioactive materials in liquid effluents released from the site (Figure 3.9-1) shall be limited:

      1. During any calendar quarter to $\leq$3.0 mrem to the total body and to $\leq$ 10 mrem to any organ, and

      2. During any calendar year to $\leq$6 mrem to the total body and to $\leq$20 mrem to any organ.

b. With the quantity of radioactive material in any of the above listed tanks exceeding the limit in (a) above, immediately suspend all additions of radioactive materials to the tank and within 48 hours reduce the tank contents to within the limit.

B. Gaseous Effluents

1. Dose Rate

a. The dose rate at any time due to radioactive materials released in gaseous effluents from the site (Figure 3.9-2) shall be limited to the following values:

1. The dose rate limit for noble gases shall be $\leq$ 500 mrem/year to the total body and $\leq$ 3000 mrem/year to the skin, and

2. The dose rate limit for I-131, tritium, and radioactive particulates with half-lives greater than eight days shall be $\leq$ 1500 mrem/year to any organ

b. With the dose rate(s) exceeding the limits in (a) above, immediately decrease the release rate to within acceptable limits.

2. Dose from Noble Gases

a. The air dose in unrestricted areas due to noble gases released in gaseous effluents from the site (Figure 3.9-2) shall be limited to the following values:

1. During any calendar quarter, to $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation, and

2. During any calendar year, to $\leq$ 20 mrad for gamma radiation and $\leq$ 40 mrad for beta radiation.

b. With the calculated air dose from radioactive noble gases in gaseous effluent exceeding any of the above limits, within 30 days submit to the Commission a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective action(s) taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

TABLE TS.3.9-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release			
a. Liquid Radwaste Effluent Line	1	During releases	1
b. Steam Generator Blowdown Effluent Line	1/Unit	During releases	2
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	During releases requiring throttling of flow	4
b. Steam Generator Blowdown Flow	1/Gen	During releases	4
3. Continuous Composite Samplers			
a. Each Turbine Building Sump Effluent Line	1/Unit	During releases	3
4. Discharge Canal Monitor	1	At all times	3
5. Tank Level Monitor			
a. Condensate Storage Tanks	1/Unit	When tanks are in use	5
b. Temporary Outdoor Tanks Holding Radioactive Liquid	1/Tank	When tanks are in use	5
6. Discharge Canal Flow System (Daily determination and following changes in flow)	NA	At all times	

TABLE TS.3.9-2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Waste Gas Holdup System Explosive Gas (Oxygen) Monitors	2	During system operation	2
2. Effluent Release Points (Unit No. 1 Reactor Bldg, Unit No. 1 Aux Bldg, Unit No. 2 Reactor Bldg, Unit No. 2 Aux Bldg, Spent Fuel Pool, Radwaste Bldg)			
a. Noble Gas Activity Monitor*	1	During releases	4, 5, 7
b. Iodine Sampler Cartridge	1	During releases	3
c. Particulate Sampler Filter	1	During releases	3
d. Sampler Flow Integrator	1	During releases	1
3. Air Ejector Noble Gas Monitors (Each Unit)	1	During power operation	6

*Noble gas activity monitors providing automatic termination of releases (except the Radwaste Building which has no automatic isolation function).

3.12 SNUBBERS

Applicability

Applies to the operability of safety related snubbers.

Objective

To define those conditions of snubber operability necessary to assure safe reactor operation.

Specification

A. Except as permitted below, all safety related snubbers shall be operable above Cold Shutdown. Snubbers may be inoperable in Cold Shutdown and Refueling Shutdown whenever the supported system is not required to be Operable.

B. With one or more snubbers made or found to be inoperable for any reason when Operability is required, within 72 hours:

1. Replace or restore the inoperable snubbers to Operable status and perform an engineering evaluation per Specification 4.13.E on the supported component(s), or

2. Declare the supported system inoperable and take the action required by the Technical Specifications for inoperability of that system.

Basis

All snubbers are required to be Operable above Cold Shutdown to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

3.14 FIRE DETECTION AND PROTECTION SYSTEMS

Applicability

Applies to instrumentation and plant systems used for fire detection and protection of the nuclear safety-related structures, systems, and components of the plant.

Objective

To insure that the structures, systems, and components of the plant important to nuclear safety are protected from fire damage.

Specification

A. Fire Detection Instrumentation

1. Except as specified below, the minimum fire detection instrumentation for each fire detection zone shown in Table 3.14-1 shall be operable whenever equipment in that fire detection zone is required to be operable. Fire detection instruments located within containment are not required to be operable during the performance of Type A containment leakage rate tests.

2. If Specification 3.14.A.1 cannot be met:

   a. Within one hour, establish a fire watch patrol to inspect the zone with the inoperable instruments at least once per hour. Fire zones located inside primary containment are exempt from this requirement when containment integrity is required.

   b. Restore the inoperable instruments to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the malfunction and the plans for restoring the instruments to operable status.

B. Fire Suppression Water System

1. Except as specified in 3.14.B.2 or 3.14.B.3 below, the system shall be operable at all times with:

   a. The following pumps, including automatic initiation logic, operable and capable of delivering at least 2000 gpm at a discharge pressure of 108 psig.

      1. Diesel-driven fire pump
      2. Motor-driven fire pump
      3. Screen wash pump

b. An operable flow path capable of taking suction from the river and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant valves and the first valve ahead of each deluge valve, hose station, or sprinkler system required to be operable.

2. With one or two of the pumps required by Specification 3.14.B.1.a inoperable, restore the inoperable equipment to operable status within seven days or provide a special report to the Commission within 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in the Fire Suppression Water system. With an inoperable pump, perform the surveillance required by Specification 4.16.B.2.

3. With the fire suppression water system otherwise inoperable:

   a. Establish a backup Fire Suppression Water System within 24 hours.

   b. Provide a special report to the Commission within 30 days outlining the actions taken and the plans and schedule for restoring the system to operable status.

   c. If Specification 3.14.B.3.a cannot be met, the reactors shall be placed in hot standby within 6 hours and in cold shutdown within 30 hours.

C. Spray and Sprinkler Systems

1. Whenever equipment protected by the following spray and sprinkler systems is required to be operable, the spray and sprinkler system shall be operable:

   a. Auxiliary Feed Pump Room WP-10

   b. Diesel Generator Areas PA-1

   c. Unit No. 1 Electrical Penetration Area PA-3

   d. Unit No. 1 Electrical Penetration Area PA-4

   e. Unit No. 2 Electrical Penetration Area PA-6

   f. Unit No. 2 Electrical Penetration Area PA-7

   g. Screenhouse PA-9

2. If Specification 3.14.C.1 cannot be met, a continuous fire watch with backup fire suppression equipment shall be established within one hour. Restore inoperable spray and sprinkler systems to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of inoperability and the plans for restoring the system to operable status.

D. Carbon Dioxide System

1. Except as specified in 3.14.D.3 below, the CO_2 system protecting the relay and cable spreading room area shall be operable with a minimum level of 60% in the CO_2 storage tank.

2. During those periods when the relay and cable spreading room area is normally occupied, automatic initiation of the CO_2 system may be bypassed. During those periods when the area is normally unoccupied, the CO_2 system shall be capable of automatic initiation unless there are personnel actually in the area.

3. If specification 3.14.D.1 cannot be met, a continuous fire watch with backup fire suppression equipment shall be stationed in the relay and cable spreading room within one hour. Restore the system to operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of inoperability and the plans for restoring the system to operable status.

F. Fire Hose Stations

1. Whenever equipment protected by hose stations in the following areas is required to be operable, the hose station(s) protecting that area shall be operable:

   a. Diesel Generator Rooms
   b. Safety Related Switchgear Rooms
   c. Safety Related Areas of Screenhouse
   d. Auxiliary Building
   e. Control Room
   f. Relay & Cable Spreading Room
   g. Battery Rooms
   h. Auxiliary Feed Pump Room

2. If Specification 3.14.E.1 cannot be met, within one hour hoses supplied from operable hose stations shall be made available for routing to each area with an inoperable hose station.

   Restore the inoperable hose station(s) to Operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the stations to Operable status.

F. Yard Hydrant Hose Houses

1. Whenever equipment in the following buildings is required to be operable, the yard hydrant hose houses in the main yard loop adjacent to each building shall be operable:

   a. Unit No. 1 Reactor Building

   b. Unit No. 2 Reactor Building

   c. Turbine Building

   d. Auxiliary Building

   e. Screen House

2. If Specification 3.14.F.1 cannot be met, within one hour have sufficient additional lengths of 2-1/2 inch diameter hose located in adjacent operable yard hydrant hose house(s) to provide service to the unprotected area(s).

   Restore the yard hydrant hose house(s) to Operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the houses to Operable status.

G. Penetration Fire Barriers

1. All penetration fire barriers in fire area boundaries protecting equipment required to be operable shall be operable.

2. If Specification 3.14.G.1 cannot be met, a continuous fire watch shall be established on at least one side of the affected penetration(s) within one hour.

   Restore the inoperable penetration fire barriers to Operable status within 14 days or submit a special report to the Commission within 30 days outlining the cause of the inoperability and the plans and schedule for restoring the barriers to Operable status.

3.5 EVENT MONITORING INSTRUMENTATION

Applicability

Applies to plant instrumentation which does not perform a protective function, but which provides information to monitor and assess important parameters during and following an accident.

Objectives

To ensure that sufficient information is available to operators to determine the effects of and determine the course of an accident to the extent required to carry out required manual actions.

Specification

A. Process Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-1 shall be Operable.

2. With the number of Operable event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-1, either restore the inoperable channels to Operable status within seven days, or be in at least Hot Shutdown within the next 12 hours.

3. With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table TS.3.15-1, either restore the minimum number of channels to Operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

B. Radiation Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-2 shall be Operable.

2. With the number of Operable event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-2, either restore the inoperable channels to Operable status within seven days, or prepare and submit a special report to the Commission within 30 days outlining the action taken, the cause of the inoperability, the plans and the schedule for restoring the system to Operable status.

3. With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirement of Table TS.3.15-2, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in 2 above.

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

	Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
3.	WATERBORNE (Continued)			
	d. Sediment from shoreline	One sample upstream of plant, one sample downstream of plant, and one from shoreline of recreational area.	Semiannually	Gamma isotopic analysis of each sample
4.	INGESTION			
	a. Milk	One sample from dairy farm having highest D/Q, one sample from each of three dairy farms calculated to have doses from I-131 > 1 mRem/yr, and one sample from 10-20 miles	Monthly or biweekly if animals are on pasture	Gamma isotopic and I-131 analysis of each sample
	b. Fish and Invertebrates	One sample of one game specie of fish located upstream and downstream of the plant site One sample of Invertebrates upstream and downstream of the plant site	Semiannually	Gamma isotopic analysis on each sample (edible portion only on fish)

**Sample locations are given on the figure and table in the ODCM

TABLE TS.4.10-1
(Page 3 of 4)

D. Tests resulting in 0.005 microcuries or more of removable contamination on the test sample shall be reported to the Commission on an annual basis.

E. Plant operating records shall be made as follows:

1. An inventory of licensed radioactive materials in possession shall be maintained current at all times.
2. The following records shall be retained for 2 years:
    a. Test results in microcuries, for tests performed pursuant to TS 4.11.
    b. Record of annual physical inventory verifying accountability of sources on record.

Bases

Licensee's program, facilities, personnel, and procedures for safe storage, handling, and use of sealed sources containing radioactive materials is described in FSAR Section 11.4. The surveillance program described in this specification is a part of licensee's program to detect and control contamination of areas in the plant by such radioactive materials. Small quantities of byproduct materials are exempt for licensing by 10 CFR 30.18 and therefore are exempt from leakage tests in this specification. Inhalation or ingestion of such small quantities of byproduct materials from a sealed source would result in less than one maximum permissible body burden for total body irradiation. Sources containing less than 0.1 microcurie of plutonium are exempt from leakage tests by 10 CFR 70.39(c) and therefore such quantities of special nuclear materials (including alpha emitters) are exempt from leakage tests in this specification. The acceptance criteria of less than 0.005 microcuries on the test sample is also based on 10 CFR 70.39(c).

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table TS.4.12-1.

E. Reports

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging, the number of tubes plugged in each steam generator shall be submitted in a special report to the Commission within 15 days.

2. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4.13 SNUBBERS

Applicability

Applies to periodic testing and surveillance requirements of safety related hydraulic snubbers.

Objective

To verify the integrity and operability of hydraulic snubbers.

Specification

The following surveillance requirements apply to all safety related snubbers. These requirements augment the inspections required by Section XI of the ASME Code.

A. Visual Inspection of snubbers shall be conducted in accordance with the following schedule:

No. of Snubbers Found Inoperable per Inspection Period	Next Required Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

B. Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, (2) attachments to the supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspection may be determined Operable for the purpose of establishing the next visual inspection interval by:

a. Clearly establishing the cause of the rejection for that particular snubber and for others that may be generically susceptible; and

b. Functionally testing the affected snubber in the as-found condition and finding it operable per Specification 4.13.D.

However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be considered inoperable for purposes of establishing the next visual inspection interval. All hydraulic snubbers connected to an inoperable common hydraulic fluid reservoir shall be considered as inoperable snubbers.

C. Except as specified below, functional testing of snubbers shall be conducted at least once per 18 months during cold shutdown. Ten percent of the total of each type snubber shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria in Specification 4.13.D below, an additional ten percent of that type of snubber shall be functionally tested until no more failures are found or all snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of the snubbers. Twenty-five percent of the sample shall include snubbers from the following three categories.

a. The first snubber away from a reactor vessel nozzle

b. Snubbers within five feet of heavy equipment (valve, pump, turbine, motor, etc.)

c. Snubbers within ten feet of the discharge of a safety/relief valve

Snubbers identified as "High Radiation Area" or "Difficult to Remove" are exempt from functional testing provided a justifiable basis for exemption is presented for Commission review; snubber life testing is performed to qualify snubber operability for all design conditions; or snubbers of the same type, configuration, and similar service have been tested for a ten year period and no failures have occurred. In such exempt cases, a qualitative test report shall be on file to substitute for the required functional testing.

In addition to the regular sample and specified re-sampling, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber, if it is repaired and installed in another position, and the spare snubber shall be retested.

If any snubber selected for functional testing either fails to lockup or fails to move (i.e., frozen in place) the cause shall be evaluated and all snubbers subject to the same defect shall be functionally tested. This testing is in addition to the regular sample and specified re-samples.

D. Hydraulic snubber functional tests shall verify that:

a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

E. An engineering evaluation shall be performed for all components supported by inoperable snubbers. The purpose of this engineering evaluation shall be to determine if the components were adversely affected by the inoperable snubber(s) to ensure that the components remain capable of meeting the designed service.

F. The installation and maintenance records for each snubber shall be reviewed at least once every 18 months to verify that the indicated service life will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement, or reconditioning shall be indicated in the records.

Basis

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

TABLE TS.4.17-1 - RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Channel Check Frequency (4)	Source Check Frequency	Functional Test Frequency	Calibration Frequency
Liquid Radwaste Effluent Line Gross Radioactivity Monitor	Daily during releases	Prior to each release	Quarterly(1)	At least once every 18 months (3)
Liquid Radwaste Effluent Line Flow Instrument	Daily during releases	--	--	At least once every 18 months
Steam Generator Blowdown Gross Radioactivity Monitors	Daily during releases	Monthly	Quarterly(1)	At least once every 18 months(3)
Steam Generator Blowdown Flow	Daily during releases	--	--	At least once every 18 months
Turbine Building Sump Continuous Composite Samplers	Daily during releases (Includes sample volume check)	--	--	At least once every 18 months
Discharge Canal Monitor	Daily during releases	Monthly	Quarterly(2)	At least once every 18 months(3)
Discharge Canal Flow Instruments	Daily during releases	--	--	At least once every 18 months
Condensate Storage Tank Level Monitors	Daily	--	Quarterly	At least once every 18 months
Level Monitors for Temporary Outdoor Tanks Holding Radioactive Liquid	Daily when in use	--	Quarterly when in use	At least once every 18 months when in use

TABLE TS.4.17-2 - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Channel Check Frequency (4)	Source Check Frequency	Functional Test Frequency	Calibration Frequency
Waste Gas Holdup System Explosive Gas (Oxygen) Monitors	Daily during system operation	--	Monthly(2)	Quarterly(5)
Effluent Release Points (Unit No. 1 Reactor Bldg, Unit No. 1 Aux Bldg, Unit No. 2 Reactor Bldg, Unit No. 2 Aux Bldg, Spent Fuel Pool, Radwaste Bldg)				
Noble Gas Activity Monitor (4) (Except Radwaste Building)	Daily during releases	Monthly*	Quarterly(1)	At least once every 18 months (3)
Noble Gas Activity Monitor Radwaste Building (4)	Daily during releases	Monthly	Quarterly(2)	At least once every 18 months (3)
Iodine and Particulate Samplers	Weekly	--	--	--
Sampler Flow Rate Monitor	Weekly	--	--	At least once every 18 months
Air Ejector Noble Gas Monitors (Each Unit)	Daily during releases	Monthly	Quarterly(2)	At least once every 18 months (3)

*A source check of the applicable noble gas monitor shall be conducted prior to each waste gas decay tank or containment purge release.

DPR-42 - Amendment No. 26, 46, 73
DPR-60 - Amendment No. 20, 40, 66

TABLE TS.6.1-1

MINIMUM SHIFT CREW COMPOSITION (Note 1 and 3)

CATEGORY	BOTH UNITS IN COLD SHUTDOWN OR REFUELING SHUTDOWN	ONE UNIT IN COLD SHUTDOWN OR REFUELING SHUTDOWN AND ONE UNIT ABOVE COLD SHUTDOWN	BOTH UNITS ABOVE COLD SHUTDOWN
No. Licensed Senior Operators (LSO)	2 (Note 2)	2 (Notes 2, 4)	2 (Note 4)
Total No. Licensed Operators (LSO & LO)	4	4	5
Total No. Licensed & Unlicensed Operators	6	7	8
Shift Technical Advisor	0	1	1

NOTES:

1. Shift crew composition may be one less than the minimum requirements for a period of time not to exceed two hours in order to accommodate an unexpected absence of one duty shift crew member provided immediate action is taken to restore the shift crew composition to within the minimum requirements specified.

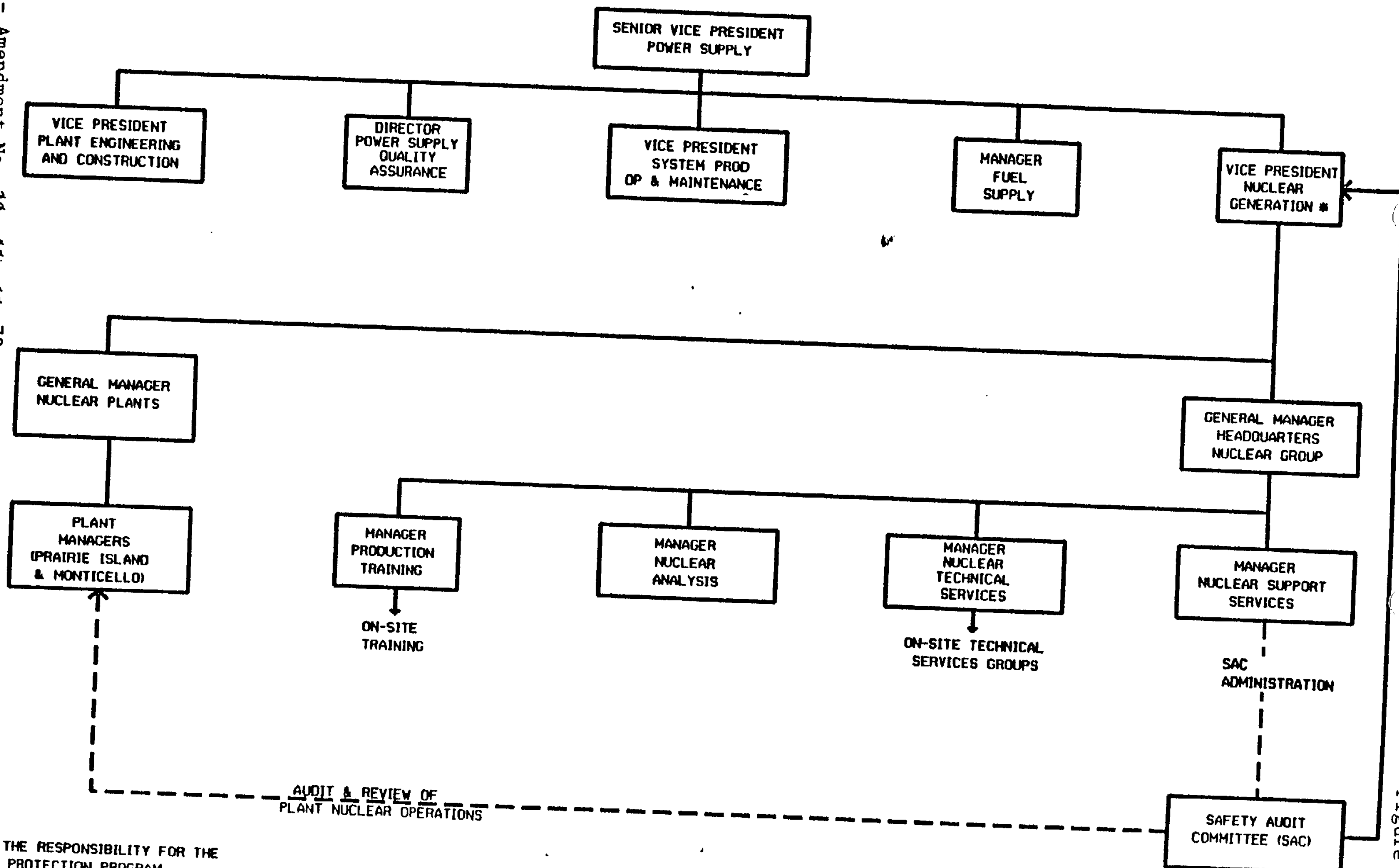
2. Does not include the licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operatons.

3. Each LSO and LO shall be licensed on each unit.

4. One LSO shall be in the control room at all times when a reactor is above cold shutdown.

Unit 1 - Amendment No. 39, 49, 61, 73

Unit 2 - Amendment No. 33, 43, 55,66



* HAS THE RESPONSIBILITY FOR THE FIRE PROTECTION PROGRAM

FIGURE TS.6.1-1 NSP CORPORATION ORGANIZATION RELATIONSHIP TO ON-SITE OPERATING ORGANIZATIONS

CAD ORTS611

Figure TS.6.1-1

6.2 Review and Audit

Organizational units for the review and audit of facility operations shall be constituted and have the responsibilities and authorities outlined below:

A. Safety Audit Committee (SAC)

The Safety Audit Committee provides the independent review of plant operations from a nuclear safety standpoint. Audits of plant operation are conducted under the cognizance of the SAC.

1. Membership

   a. The SAC shall consist of at least five (5) persons.

   b. The SAC chairman shall be an NSP representative, not having line responsibility for plant operation, appointed by the Vice President Nuclear Generation. Other SAC members shall be appointed by the Vice President Nuclear Generation or by such other person as he may designate. The Chairman shall appoint a Vice Chairman from the SAC membership to act in his absence.

   c. No more than two members of the SAC shall be from groups holding line responsibility for operation of the plant.

   d. A SAC member may appoint an alternate to serve in his absence, with concurrence of the Chairman. No more than one alternate shall serve on the SAC at any one time. The alternate member shall have voting rights.

2. Qualifications

   a. The SAC members should collectively have the capability required to review activities in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, quality assurance practices, and other appropriate fields associated with the unique characteristics of the nuclear power plant.

f. Investigation of all Reportable Events and events requiring Special Reports to the Commission.

g. Revisions to the Facility Emergency Plan, Facility Security Plan, and the Fire Protection Program.

h. Operations Committee minutes to determine if matters considered by that Committee involve unreviewed or unresolved safety questions.

i. Other nuclear safety matters referred to the SAC by the Operations Committee, plant management or company management.

j. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures systems, or components.

k. Reports of special inspections and audits conducted in accordance with specification 6.3.

l. Changes to the Offsite Dose Calculation Manual (ODCM).

m. Review of investigative reports of unplanned releases of radioactive material to the environs.

6. Audit - The operation of the nuclear power plant shall be audited formally under the cognizance of the SAC to assure safe facility operation.

a. Audits of selected aspects of plant operation, as delineated in Paragraph 4.4 of ANSI N18.7-1972, shall be performed with a frequency commensurate with their nuclear safety significance and in a manner to assure that an audit of all nuclear safety-related activities is completed within a period of two years. The audits shall be performed in accordance with appropriate written instructions and procedures.

b. Audits of aspects of plant radioactive effluent treatment and radiological environmental monitoring shall be performed as follows:

1. Implementation of the Offsite Dose Calculation Manual at least once every two years.

2. Implementation of the Process Control Program for solidification of radioactive wastes at least once every two years.

3. The Radiological Environmental Monitoring Program and the results thereof, including quality controls, at least once every year.

c. Periodic review of the audit program should be performed by the SAC at least twice a year to assure its adequacy.

d. Written reports of the audits shall be reviewed by the Vice President Nuclear Generation, by the SAC at a scheduled meeting, and by members of management having responsibility in the areas audited.

B. Operations Committee (OC)

1. Membership

   The Operations Committee shall consist of at least six (6) members drawn from the key supervisors of the onsite staff. The Plant Manager shall serve as Chairman of the OC and shall appoint a Vice Chairman from the OC membership to act in his absence.

2. Meeting Frequency

   The Operations Committee will meet on call by the Chairman or as requested by individual members and at least monthly.

3. Quorum

   A majority of the permanent members, including the Chairman or Vice Chairman

4. Responsibilities - The following subjects shall be reviewed by the Operations Committee:

   a. Proposed tests and experiments and their results.

   b. Modifications to plant systems or equipment as described in the Updated Safety Analysis Report and having nuclear safety significance or which involve an unreviewed safety question as defined in Paragraph 50.59 (c), Part 50, Title 10, Code of Federal Regulations.

   c. Proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that will affect nuclear safety as determined by the Plant Manager.

   d. Proposed changes to the Technical Specifications or operating licenses.

   e. All reported or suspected violations of Technical Specifications, operating license requirements, administrative procedures, operating procedures. Results of investigations, including evaluation and recommendations to prevent recurrence will be reported in writing to the Vice President Nuclear Generation and to the Chairman of the Safety Audit Committee.

f. Investigations of all Reportable Events and events requiring Special Reports to the Commission.

g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with offsite support groups.

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan, and the Security Plan, shall be reviewed initially and periodically with a frequency commensurate with their safety significance but at an interval of not more than two years.

i. Special reviews and investigations, as requested by the Safety Audit Committee.

j. Review of investigative reports of unplanned releases of radioactive material to the environs.

k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

The OC shall be advisory to the Plant Manager. In the event of a disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the General Manager Nuclear Plants and the Chairman of the SAC for review.

6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Manager Nuclear Plants and others designated by the OC Chairman or Vice Chairman.

7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group

b. Content and method of submission of presentations to the Operations Committee

c. Mechanism for scheduling meetings

d. Provision for meeting agenda

7. Report of Safety and Relief Valve Failures and Challenges. An annual report of pressurizer safety and relief valve failures and challenges shall be submitted prior to March 1 of each year.

B. Reportable Events

The following actions shall be taken for Reportable Events:

a. The Commission shall be notified by a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each Reportable Event shall be reviewed by the Operations Committee and the results of this review shall be submitted to the Safety Audit Committee and the Vice President Nuclear Generation.

C. Environmental Reports

The reports listed below shall be submitted to the Administrator of the appropriate Regional NRC Office or his designate:

1. Annual Radiation Environmental Monitoring Report

(a) Annual Radiation Environmental Monitoring Reports covering the operation of the program during the previous calendar year shall be submitted prior to May 1 of each year.

(b) The Annual Radiation Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 4.10.B.1. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

(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.

(d) The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 4.10.C.1.

2. Environmental Special Reports

(a) When radioactivity levels in samples exceed limits specified in Table 4.10-3, an Environmental Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 day period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Environmental Special Report as soon as practicable.

3. Other Environmental Reports (non-radiological, non-aquatic)

Written reports for the following items shall be submitted to the appropriate NRC Regional Administrator:

a. Environmental events that indicate or could result in a significant environmental impact causually related to plant operation. The following are examples: excessive bird impaction; onsite plant or animal disease outbreaks; unusual mortality of any species protected by the Endangered Species Act of 1973; or increase in nuisance organisms or conditions. This report shall be submitted within 30 days of the event and shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

b. Proposed changes, test or experiments which may result in a significant increase in any adverse environmental impact which was not previously reviewed or evaluated in the Final Environmental Statement or supplements thereto. This report shall include an evaluation of the environmental impact of the proposed activity and shall be submitted 30 days prior to implementing the proposed change, test or experiment.

D. Special Reports

Unless otherwise indicated, special reports required by the Technical Specification shall be submitted to the appropriate NRC Regional Administrator within the time period specified for each report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 73 AND 66

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

NORTHERN STATES POWER COMPANY

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

DOCKET NOS. 50-282 AND 50-306

Introduction

By letter dated July 11, 1984 as supplemented April 26, 1985, Northern States Power Company (NSP), the licensee, requested amendments to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 (PINGP). The requested amendments proposed changes to the Technical Specifications (TS) in the following areas:

1. TS 1-1, (and throughout sections of TSs) - Administrative changes dealing with reporting requirements as defined in Generic Letter (GL) 83-43 and 10 CFR 50.73

2. TS i thru TS viii - Administrative changes revising the Table of Contents, detailing additional listings that result in easy access for locating items in the TS.

3. TS 1.P.3, TS 3.6.A.4 and TS 3.8.A.4 - A change in the 2000 PPM Boron requirements to achieve consistency with the standard technical specification requirements for the reactor coolant system during various refueling conditions.

4. TS 4.11D - An administrative change involving the deletion of the reference to TS section 6.7 for the reporting requirements related to the radioactive source leakage test.

5. TS 6.1-1 table - An addition to the TS is proposed to meet a new requirement of 10 CFR 50.54(m)(2) for having one Senior Reactor Operator (SRO) in the control room at all times when the reactor is above cold shutdown.

6. TS 3.12-1 table and TS 4.13-1 - The proposed changes implement the recommendation of Generic Letter 84-13 dealing with the deletion of the detailed listing of the safety related snubbers.

7. TS 3.3B1c - The proposed change is administrative in nature and is related to the minimum volume and concentration of the sodium hydroxide spray additive tank.

8. TS 3.9-1 - The proposed change modifies the requirement for measuring the plant discharge flow rate which would make continuous flow rate measurement unnecessary.

9. TS 3.9-2 table, TS 4.17-1 table - The proposed change involves the deletion of unnecessary tests of the radioactive effluent monitoring instrumentation.

10. TS 4.10-1 table - The proposed change involves minor wording changes regarding the radiation environmental monitoring program sample collection and analysis to achieve consistency with the Standard Technical Specifications for PWRs (NUREG-0473 Revision 2).

11. TS 6.2 and Figure TS 6.1-1 - The proposed change is administrative in nature in that it involves a title change of the Director of Nuclear Generation.

By letter dated January 21, 1985, the licensee withdrew the portion of the license amendment request associated with new peaking factor limits for the Exxon fuel assemblies. The revised peaking factor limits will not be needed for the current cycles for both units. In addition, these peaking factor limits which are applicable to EXXON fuel would not be needed for future cycles since the licensee is scheduled to change fuel vendors starting with cycle 11 on both units. On this basis, the staff stopped the review of this change and the technical specifications are not amended to reflect this item.

Discussion and Evaluation

1. TS 1-1 (and throughout applicable section of TS Reporting Requirements

By Generic Letter 83-43, we requested all licensees and applicants to implement changes to the technical specifications (TS) to reflect the requirement of 10 CFR 50.73: "Since paragraph (g) of Section 50.73 specifically states that 'the requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications,' the reporting requirements incorporated into the "Administrative Controls" section of your facility's technical specifications may require modification." The licensee responded to our request by submitting the proposed changes to the TS to reflect the new requirements of 10 CFR 50.73. The licensee, using model technical specifications as guidance, added a definition for "Reportable Events", changed the phrases "reportable occurrence report," "prompt notification with written followup" and other similar phrases. The staff has reviewed the licensee's proposed changes and finds that the proposed changes agree with the guidance given in GL 83-43. On this basis, we find the proposed changes to the reporting requirements acceptable.

2. TS i thru TS viii Table of Contents

The licensee proposed to change the Table of Contents of the TS, giving a more detailed breakdown of the various sections. We agree with the licensee that the detailed listing of these items would make it easier to locate the various subjects throughout the body of the TS. This change is administrative in nature and has no effect on the limiting conditions of operation, the surveillance or the administrative requirements. On this basis, the staff concludes that there will be no reduction in the level of plant safety and therefore, the changes are acceptable.

3. TS 3.6 and TS 3.8 Refueling Boron Concentration

The licensee has requested that the limit on boron concentration of 2000 ppm during refueling operations be deleted. The current TS specific limits of 2000 ppm boron and a shutdown margin of 10% Δk/k. In order to eliminate any confusion as to which limit must be observed, the licensee proposed the deletion of the 2000 ppm boron concentration limit. By the letter dated April 26, 1985, the licensee confirmed that the boron concentration corresponding to a shutdown margin of 10% Δk/k is calculated as part of the reload analysis and is stated in the Startup and Operations Report and in the plant procedures. The licensee's objective of this change is to achieve consistency throughout the TS to assure that the shutdown margin requirement is not misinterpreted.

The Standard Technical Specifications permit the more restrictive of the 2000 ppm boron or a shutdown margin of 5% Δk/k to exist during refueling operations. The boron concentration necessary to ensure a shutdown margin of 10% Δk/k is equivalent to 2000 ppm boron concentration and therefore specifying either value in the technical specification will in no way reduce the safety margin during refueling operation for which the specification applies. The 5% Δk/k has been included in the proposed change to cover rare occasions when the 2000 ppm boron concentration would not be adequate to maintain the 10% Δk/k. The licensee's proposed change is inconsistent in regard to the 2000 PPM boron and 5% Δk/k limits specified in the Standard Technical Specifications. Therefore, the licensee's proposed change was modified to eliminate this inconsistency. The modifications to the proposed change were discussed with and agreed to by the licensee. The staff concludes the proposed change as modified adequately meets the intent of the Standard Technical Specification requirements and eliminates the potential of misinterpretation of the shutdown margin requirements in the TS. In addition, the proposed change adequately protects against inadvertent criticality during refueling operations. On this basis, the staff finds the proposed change will not reduce the level of plant safety and thus finds it acceptable.

4. TS 4.11 Radioactive Source Leakage Test

The licensee proposes to delete the reference to TS section TS 6.7 "reporting requirements" in describing the reporting requirements for source leak test results. This proposed change is administrative in nature and does in no way change the reporting requirement for source leak test results. A review of section 6.7 by the staff shows that the requirement of this section does not apply nor does it give any applicable guidance related to reporting the results of source leak tests. A misspelled word appearing in the basis (i.e., ingestion) also has been corrected. Based on the above, the proposed change dealing with the deletion of the reference to section TS 6.7 will not reduce or affect the level of plant safety and therefore is acceptable.

5. TS 6.1-1 table-Senior Reactor Operator (SRO) Requirements

The licensee proposed a change to Table TS 6.1-1 regarding the minimum shift crew composition to assure agreement with 10 CFR 50.54(m)(2) that became effective January 1, 1984. Specifically, the licensee proposes to add a note to the table specifying that one SRO shall be in the control room at all times when one or both reactors are above cold shutdown. This added requirement is merely having the TS agree with requirements of 10 CFR 50.54 and is an additional limiting condition of operation that increases the level of plant safety. On this basis, the staff finds the proposed change acceptable.

6. TS 3.12-1 Deletion of Snubber table

By letter dated October 28, 1981, the Commission issued Amendment Nos. 50 and 44 to Facility Operating License Nos. DPR-42 and DPR-60 that put in place the Hydraulic Snubber technical specifications. The TS were revised to incorporate the surveillance requirements as requested by our Generic Letter dated November 20, 1980. Generic Letter 84-13 dated May 3, 1984, transmitted Revision 1 of the Surveillance Requirements of the Generic Letter dated November 20, 1980, and allowed the deletion of the tabular listing of safety related snubbers provided the snubber TS is modified to specify which snubbers are required to be operable. The licensee's proposed change includes all safety related snubbers to be operable when the reactor is above cold shutdown. The safety related snubbers include all snubbers necessary to insure that the structural integrity of the reactor coolant system and all other safety related systems are maintained during and following a seismic event or other events initiating dynamic loads. Based on the above and our review of the licensee's proposed change, the licensee has adequately specified which snubbers are required to be operable and meets the intent of Generic Letter 84-13. In addition, the licensee is required to maintain plant records containing a record of the service life, installation date and maintenance history for each snubber. On this basis, the proposed change is acceptable.

7. TS 3.3B1C Spray Additive tank volume/concentration requirements

By letter dated September 1, 1983 the Commission issued Amendments Nos. 65 and 59 permitting the lowering of the Sodium Hydroxide (NaOH) concentration in the spray additive tanks of the containment spray system from 30% by weight to a range of 9% to 11% by weight, inclusive. The licensee's analysis showed that, under certain accident scenarios, the electrical equipment inside containment could be subjected to an unacceptable pH environment (i.e., pH > 10.5) if the 30% by weight NaOH concentration is maintained in the spray additive tanks. Amendment Nos. 65 and 59 to the TS still permitted the 30% by weight NaOH concentration to exist in spray additive tanks until the tanks were modified to accept the larger volume of solution for the lower NaOH concentration range (i.e., 9% to 11% by weight). This proposed change is administrative in nature in that it deletes the 30% by weight. NaOH concentration from the TS which is no longer necessary or required since the modifications to the spray additive tanks are now completed. Thus, the proposed TS change would have a single requirement for the NaOH concentration in the spray additive tanks that has been found acceptable in Amendment Nos. 65 and 59. On this basis, the staff finds the proposed deletion of 30% by weight NaOH concentration is acceptable.

8. TS 3.9-1 table-Discharge Canal Flow

9. TS 3.9-2 table, TS 4.17-1 and TS 4.17-2 Radioactive Effluent Monitoring Instrumentation Requirements

10. TS 4.10-1 table-Radiation Environmental Monitoring Program Sample Collection and analysis

Items 8, 9 and 10 are proposed changes that modify requirements related to the Radiological Effluent Technical Specifications (RETS) that were issued by the Commission on October 21, 1982 as Amendment Nos. 59 and 53. The items address the following areas:

a. Item 8 is related to a clarification of the measurement of the plant discharge flow rate frequency (i.e., continuous vs daily).

b. Item 9 is related to deleting unnecessary testing of Radiation Effluent Monitoring instrumentation; and;

c. Item 10 deals with minor wording changes to achieve consistency with the Standard Technical Specifications for PWRs (NUREG-0473, Revision 2).

All of these changes in items 8, 9 and 10 were reviewed by the staff. The staff finds that the modifications proposed therein meet the intent of the NRC staff's RETS for PWRs, NUREG-0473 Revision 2, (February 1, 1980). The changes proposed in these items will not remove or relax any existing requirements related to the probability or consequences of accidents previously considered and do not involve a significant hazards consideration. In addition, the proposed changes will not remove or relax any existing requirements needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. On this basis, the staff finds the proposed changes acceptable.

11. TS 6.1-1 table, TS 6.2 and TS 6.7 title change

The licensee proposed the title of "Director of Nuclear Generation" be changed to "Vice President Nuclear Generation. The administrative sections of the TS reflect the change by modifying the title of the Director of Nuclear Generation to Vice President Nuclear Generation. The title change would make the title appearing in the TS agree with the title existing in the actual plant organization. This change is administrative in nature and has no effect on the management function, authority or responsibility of the position identified throughout the administrative sections of the TS. On this basis, the staff concludes that the change will not reduce the level of plant safety and, therefore, is acceptable.

Environmental Consideration

The part of the amendment dealing with clarifications and elimination of unnecessary testing in the radiation environmental monitoring program technical specifications involves an administrative change. Accordingly, this part of the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). The remaining parts of the amendments involve a change in the installation or use of a facility component located within the restricted area or a change to a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the remaining parts of the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 25, 1985

Principal Contributors:
D. C. Di Ianni
M. Chatterton
W. Meinke
C. Willis