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

February 2, 1983

Docket Nos. 50-282
and 50-306

Posted
Amdt. 61
to DPR-42
(See Correction letter
of 2-15-83)

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. 61 and 55 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 September 14, 1982.

The amendments revise the Technical Specifications (TS) concerned with Steam Generator Tube Surveillance, Steam Exclusion System, Auxiliary Feedwater System reporting requirements, Administrative Controls, and correct typographical errors.

The amendments revise the TS concerned with all of the above areas except for item 2 "TS 4.12.B.4. Steam Generator Tube Surveillance" for which you have not furnished an adequate basis.

A copy of the Safety Evaluation and of the Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 61 to DPR-42
2. Amendment No. 55 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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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. 61
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 September 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 61 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 2, 1983

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. 55
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 September 14, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 55 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Within 90 days after the effective date of this amendment, or such later time as the Commission may specify, the Licensee shall satisfy any applicable requirement of P.L. 97-425 related to pursuing an agreement with the Secretary of Energy for the disposal of high-level radioactive waste and spent nuclear fuel.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: February 2, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. 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 change.

Remove	Insert
TS 3.1-3	TS 3.1-3
TS 3.3-1	TS 3.3-1
TS 3.3-2	TS 3.3-2
TS 3.3-5A	TS 3.3-5A
TS 3.4-2	TS 3.4-2
TS 3.4-3	TS 3.4-3
Table TS.4.1-1 (Pg 2 of 5)	Table TS.4.1-1 (Pg 2 of 5)
Table TS.4.1-1 (Pg 3 of 5)	Table TS.4.1-1 (Pg 3 of 5)
Table TS.4.1-1 (Pg 4 of 5)	Table TS.4.1-1 (Pg 4 of 5)
Table TS.4.1-1 (Pg 5 of 5)	Table TS.4.1-1 (Pg 5 of 5)
Table TS.4.2-1	Table TS.4.2-1
TS 4.5-2	TS 4.5-2
Table TS.4.12-1	Table TS.4.12-1
TS 5.6-2	TS 5.6-2
TS 6.1-1	TS 6.1-1
Figure TS 6.1-1 & TS 6.1-2	Figure TS 6.1-1 & TS 6.1-2
TS 6.2-1	TS 6.2-1
TS 6.2-3	TS 6.2-3
TS 6.2-4	TS 6.2-4
TS 6.2-5	TS 6.2-5
TS 6.5-1	TS 6.5-1

Basis

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the primary system volume in approximately one-half hour.

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth $\geq$50% of the 0.050-inch tube wall thickness as being unacceptable for power operation. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.[2]

Part A of the specification requires that both reactor coolant pumps be operating when the reactor is critical to provide core cooling in the event that a loss of flow occurs. In the event of the worst credible coolant flow loss (loss of both pumps from 100% power) the minimum calculated DNBR remains well above 1.30. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Critical operation, except for low power physics tests, with less than two pumps is not planned. Above 10% power, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load.[1]

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the engineered safety features.

Objective

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

Specifications

A. Safety Injection and Residual Heat Removal Systems

1. A reactor shall not be made or maintained critical nor shall it be heated or maintained above 200°F unless the following conditions are satisfied except as permitted in Specification 3.3.A.2.

   a. The refueling water tank contains not less than 200,000 gallons of water with a boron concentration of at least 1950 ppm.

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

   Operability requires:
   (1) The isolation valve is open
   (2) Volume is between 1250 and 1282.9 cubic feet of borated water
   (3) A minimum boron concentration of 1900 ppm
   (4) A nitrogen cover pressure of at least 700 psig

   c. Two safety injection pumps are operable except that pump control switches in the control room shall meet the requirements of Section 3.1.G whenever the reactor coolant system temperature is less than MPT.

   d. Two residual heat removal pumps are operable.

   e. Two residual heat exchangers are operable.

   f. Automatic valves, interlocks and piping associated with the above components and required to function during accident conditions, are operable.

   g. Manual valves in the above systems that could (if one is improperly positioned) reduce injection flow below that assumed for accident analyses, shall be blocked and tagged in the proper position for injection. RHR system valves, however, may be positioned as necessary to regulate plant heatup or cooldown rates when the reactor is subcritical. All changes in valve position shall be under direct administrative control.

d. Any redundant valve in the system required for safety injection, may be inoperable provided repairs are completed within 24 hours. Prior to initiating repairs, all valves in the system that provide redundancy shall be tested to demonstrate operability.

e. One accumulator may be inoperable for up to one hour whenever pressurizer pressure is greater than 1000 psig.

f. One safety injection system and one residual heat system may be inoperable for a time interval not to exceed 24 hours provided the redundant safety injection system and heat removal system required for functioning during accident conditions is operable.

B. <u>Containment Cooling Systems</u>

1. A reactor shall not be made or maintained critical nor shall it be heated above 200°F unless the following conditions are satisfied except as permitted by Specification 3.3.B.2.

    a. Two containment spray pumps are operable.

    b. Four fan cooler units are operable.

a. One diesel-driven cooling water pump may be inoperable for a period not to exceed seven days (total for both diesel-driven cooling water pumps during any consecutive 30 day period) provided:

(1) the operability of the other diesel-driven pump and its associated diesel generator are demonstrated immediately and at least once every 24 hours thereafter.

(2) the engineered safety features associated with the operable diesel-driven cooling water pump are operable; and

(3) both off-site power supply paths from the grid to the 4Kv emergency buses are operable.

(4) two motor-driven cooling water pumps shall be operable.

b. One of the two required motor-driven cooling water pumps may be inoperable for a period not to exceed seven days provided:

(1) the operability of both diesel-driven cooling water pumps is demonstrated immediately and at least once every 24 hours thereafter.

c. One of the two required cooling water headers may be inoperable for a period not to exceed 24 hours provided:

(1) the operability of the diesel-driven pump and the diesel generator associated with safety features on the operable header is demonstrated immediately.

(2) the horizontal motor-driven pump associated with the operable header and the vertical motor-driven pump are operable.

e. For Unit 1 operation motor operated valves MV32242 and MV32243 shall have valve position monitor lights operable and shall be locked in the open position by having the motor control center supply breakers manually locked open. For Unit 2, corresponding valve conditions shall exist.

f. Essential features including system piping, valves, and interlocks directly associated with the above components are operable.

g. Manual valves in the above systems that could (if one is improperly positioned) reduce flow below that assumed for accident analysis shall be locked in the proper position for emergency use. During power operation, changes in valve position will be under direct administrative control.

h. The condensate supply cross connect valves C-41-1 and C-41-2 to the auxiliary feedwater pumps shall be blocked and tagged open. Any changes in position of these valves shall be under direct administrative control.

3. Steam Exclusion System

a. Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be operable.

b. If one of the two redundant dampers is removed from service for testing and maintenance purposes or found inoperable, the operable redundant damper may remain open for a period not to exceed 24 hours. If after 24 hours, the inoperable damper is not returned to service, one of the two dampers shall be closed.

c. The actuation logic for one train of steam exclusion may be out of service for 24 hours provided the other train is tested and found operable prior to intiating repair of the inoperable channel.

4. Radiochemistry

The specific activity of the secondary coolant system for that reactor shall be $\leq$ 0.10 uCi/gm DOSE EQUIVALENT I-131.

B. If, during startup operation or power operation, any of the conditions of Specification 3.4.A., except as noted below for 2.a, 2.b or 4 cannot be met, startup operations shall be discontinued and if operability cannot be restored within 48 hours, the affected reactor shall be placed in the cold shutdown condition using normal operating procedures.

With regard to Specifications 2.a or 2.b, if a turbine driven AFW pump is not operable, that AFW pump shall be restored to operable status within 72 hours or the affected reactor shall be cooled to less than 350°F within the next 12 hours. If a motor driven AFW pump is not operable, that AFW pump shall be restored to operable status within 72 hours or one unit shall be cooled to less than 350°F within the next 12 hours.

If 4. is not met, the affected reactor shall be placed in hot standby within 6 hours and cold shutdown within the following 30 hours.

Basis

A reactor shutdown from power requires removal of decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten main steam safety valves have a total combined rated capability of 7,745,000 lbs/hr. The total full power steam flow is 7,094,000 lbs/hr; therefore, the ten main steam safety valves will be able to relieve the total steam flow if necessary.(1)

In the unlikely event of complete loss of offsite electrical power to either or both reactors, continued removal of decay heat would be assured by availability of either the steam-driven auxiliary feedwater pump or the motor-driven auxiliary feedwater pump associated with each reactor, and by steam discharge to the atmosphere through the main steam safety valves. One auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from one reactor. The motor-driven auxiliary feedwater pump for each reactor can be made available to the other reactor.

The minimum amount of water specified for the condensate storage tanks is sufficient to remove the decay heat generated by one reactor in the first 24 hours of shutdown. Essentially unlimited replenishment of the condensate storage supply is available from the intake structures through the cooling water system.

The two power-operated relief valves located upstream of the main steam isolation valves are required to remove decay heat and cool the reactor down following a high energy line rupture outside containment.(2) Isolation dampers are required in ventilation ducts that penetrate those rooms containing equipment needed for the accident.

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

References

(1) FSAR, Section 10.4
(2) FSAR, Appendix I

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
8.	4KV Voltage & Frequency	NA	R	M	NA	Reactor protection circuits only
8a.	RCP Breakers	NA	R	T	NA	
9.	Analog Rod Position	S(1) M(2)	R	T(2)	NA	1) With step counters 2) Rod Position Deviation Monitor Tested by updating computer bank count and comparing with analog rod position test signal
10.	Rod Position Bank Counters	S(1,2) M(3)	NA	T(3)	NA	1) With analog rod position 2) Following rod motion in excess of six inches when the computer is out of service 3) Control rod banks insertion limit monitor and control rod position deviation monitors
11.	Steam Generator Level	S	R	M	NA	
12.	Steam Generator Flow Mismatch	S	R	M	NA	
13.	Charging Flow	S	R	NA	NA	
14.	Residual Heat Removal Pump Flow	S(1)	R	NA	NA	1) When in operation
15.	Boric Acid Tank Level	D	R(1)	M(1)	NA	1) Transfer logic to Refueling Water Storage Tank

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
16.	Refueling Water Storage Tank Level	W	R	M(1)	NA	1) Functional test can be performed by bleeding transmitter
17.	Volume Control Tank	S	R	NA	NA	
18a.	Containment Pressure SI Signal	S	R	M(1)	NA	Wide Range Containment Pressure 1) Isolation Valve Signal
18b.	Containment Pressure Steam Line Isolation	S	R	M	NA	Narrow Range Containment Pressure
18c.	Containment Pressure Containment Spray	S	R	M	NA	
18d.	Annulus Pressure (Vacuum Breaker)	NA	R	R	NA	
19.	Deleted					
20.	Boric Acid Make-up Flow Channel	NA	R	NA	NA	
21.	Containment Sump Level	NA	R	R	NA	Includes Sumps A, B, and C
22.	Accumulator Level and Pressure	S	R	R	NA	
23.	Steam Generator Pressure	S	R	M	NA	
24.	Turbine First Stage Pressure	S	R	M	NA	
25.	Emergency Plan Radiation Instruments	*M	R	M	NA	Includes those named in the emergency procedure (referenced in Spec. 6.5 A.6)
26.	Protection Systems Logic Channel Testing	NA	NA	M	NA	Includes auto load sequencers

Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
27. Turbine Overspread Protection Trip Channel	NA	R	M	NA	
28. Deleted					
29. Deleted					
30. Deleted					
31. Seismic Monitors	R	R	NA	NA	Includes those reported in Item 4 of Table TS.6.7-1
32. Coolant Flow - RTD Bypass Flowmeter	S	R	M	NA	
33. CRDM Cooling Shroud Exhaust Air Temperature	S	NA	R	NA	FSAR page 3.2-56
34. Reactor Gap Exhaust Air Temperature	S	NA	R	NA	FSAR page 5.4-2
35. Post-Accident Monitoring	M	R	NA	NA	Includes all those in FSAR Table 7.7-2 and Table TS.3.15-1 not included elsewhere in this Table
36. Steam Exclusion Actuation System	W	R	M	NA	See FSAR Appendix I, Section I.14.6
37. Overpressure Mitigation System	NA	R	R	NA	Instrument Channels for PORV Control Including Overpressure Mitigation System
38. Degraded Voltage 4KV Safeguard Busses	NA	R	M	NA	
39. Loss of Voltage 4KV Safeguard Busses	NA	R	M	NA	

TABLE TS.4.1-1
(Page 5 of 5)

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
40.	Auxiliary Feedwater Pump Suction Pressure	NA	R	R	NA	
41.	Auxiliary Feedwater Pump Discharge Pressure	NA	R	R	NA	

S - Each Shift
D - Daily
W - Weekly
M - Monthly
Q - Quarterly
R - Each refueling shutdown
P - Prior to each startup if not done previous week
T - Prior to each startup following shutdown in excess of 2 days if not done in the previous 30 days
NA - Not Applicable
* - See Specification 4.1.D

SPECIAL INSERVICE INSPECTION REQUIREMENTS

Component	Method of Examination	Extent and Frequency
REACTOR COOLANT PUMPS		
1. Pump Flywheel	U.T.	An in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and key way at approx. 3 year intervals, during the refueling or maintenance shutdown coinciding with the in-service inspection schedule as required by the ASME B & PV Code Section XI.
	M.T. or P.T. U.T.	A surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approx. 10 year intervals, during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME B & PB Code Section XI. Removal of the flywheel is not required to perform these examinations.

Notes:

1. The following definitions shall apply to the inspection methods employed in Table TS.4.2-1.

   a. U.T. - Ultrasonic examination per IWA-2230.
   b. P.T. - Liquid Penetrant examination per IWA-2220.
   c. M.T. - Magnetic Particle examination per IWA-2220.

3. Containment Fan Coolers

Each fan cooler unit shall be tested during each reactor refueling shutdown to verify proper operation of all essential features including low motor speed, cooling water valves, and normal ventilation system dampers. Individual unit performance will be monitored by observing the terminal temperatures of the fan coil unit and by verifying a cooling water flow rate of greater than or equal to 900 gpm to each fan coil unit.

4. Component Cooling Water System

a. System tests shall be performed during each reactor refueling shutdown. Operation of the system will be initiated by tripping the actuation instrumentation.

b. The test will be considered satisfactory if control board indication and visual observations indicate that all components have operated satisfactorily.

5. Cooling Water System

a. System tests shall be performed at each refueling shutdown. Tests shall consist of an automatic start of each diesel engine and automatic operation of valves required to mitigate accidents including those valves that isolate non-essential equipment from the system. Operation of the system will be initiated by a simulated accident signal to the actuation instrumentation. The tests will be considered satisfactory if control board indication and visual observations indicate that all components have operated satisfactorily and if cooling water flow paths required for accident mitigation have been established.

b. Each diesel engine shall be inspected at each refueling shutdown.

B. Component Tests

1. Pumps

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

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

TABLE TS.4.12-1

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Prompt notification to NRC.	N/A	N/A

S=3%; When two steam generators are inspected during that outage.
S=6%; When one steam generator is inspected during that outage.

The spent fuel pool has a reinforced concrete bottom slab nearly 6 feet thick and has been designed to minimize loss of water due to a dropped cask accident. Such water loss, if it did occur, would be from the smaller of the two compartments, leaving the stored fuel in the larger compartment still covered with water. Piping to the pool is arranged so that failure of any pipe cannot drain more than 3 feet of water from the pool. This leaves a margin of 22 feet of water above the tops of the stored fuel assemblies.

C. Fuel Handling

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

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

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

The spent fuel cask will be lowered 66 feet from the auxiliary building to the railroad car for offsite transportation. Specification 3.8 will limit this loading operation so that if the cask drops 66 feet, there will not be a significant release of fission products from the fuel in the cask.

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

A. The Plant Manager has the overall full-time onsite responsibility for safe operation of the facility. During periods when the Plant Manager is unavailable, he may delegate this responsibiliity to other qualified supervisory personnel.

B. The Northern States Power corporate organizational structure relating to the operation of this plant is shown on Figure TS.6.1-1.

C. The functional organization for operation of the plant shall be as shown in Figure TS.6.1-2 and:

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown on Table TS.6.1-1.

2. For each reactor that contains fuel: a licensed operator in the control room.

3. At least two licensed operators shall be present in the control room during a reactor startup, a scheduled reactor shutdown, and during recovery from a reactor trip. These operators are in addition to those required for the other reactor.

4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor.

5. All refueling operations shall be directly supervised by a licensed Senior Reactor Operator or a Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6. A fire brigade of at least five members shall be maintained on site at all times.* The fire brigade shall not include the six members of the minimum shift crew for safe shutdown of the reactors.

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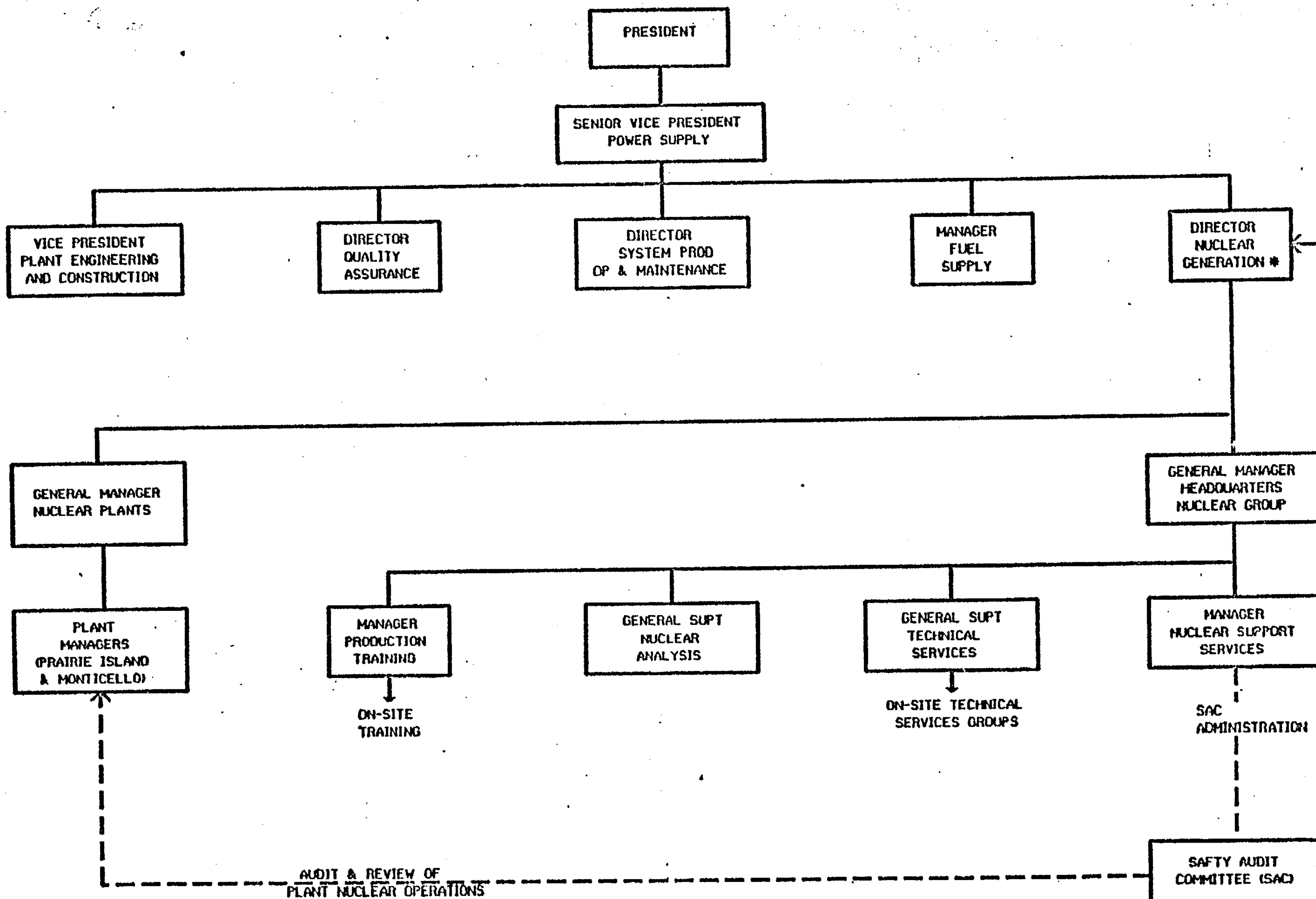
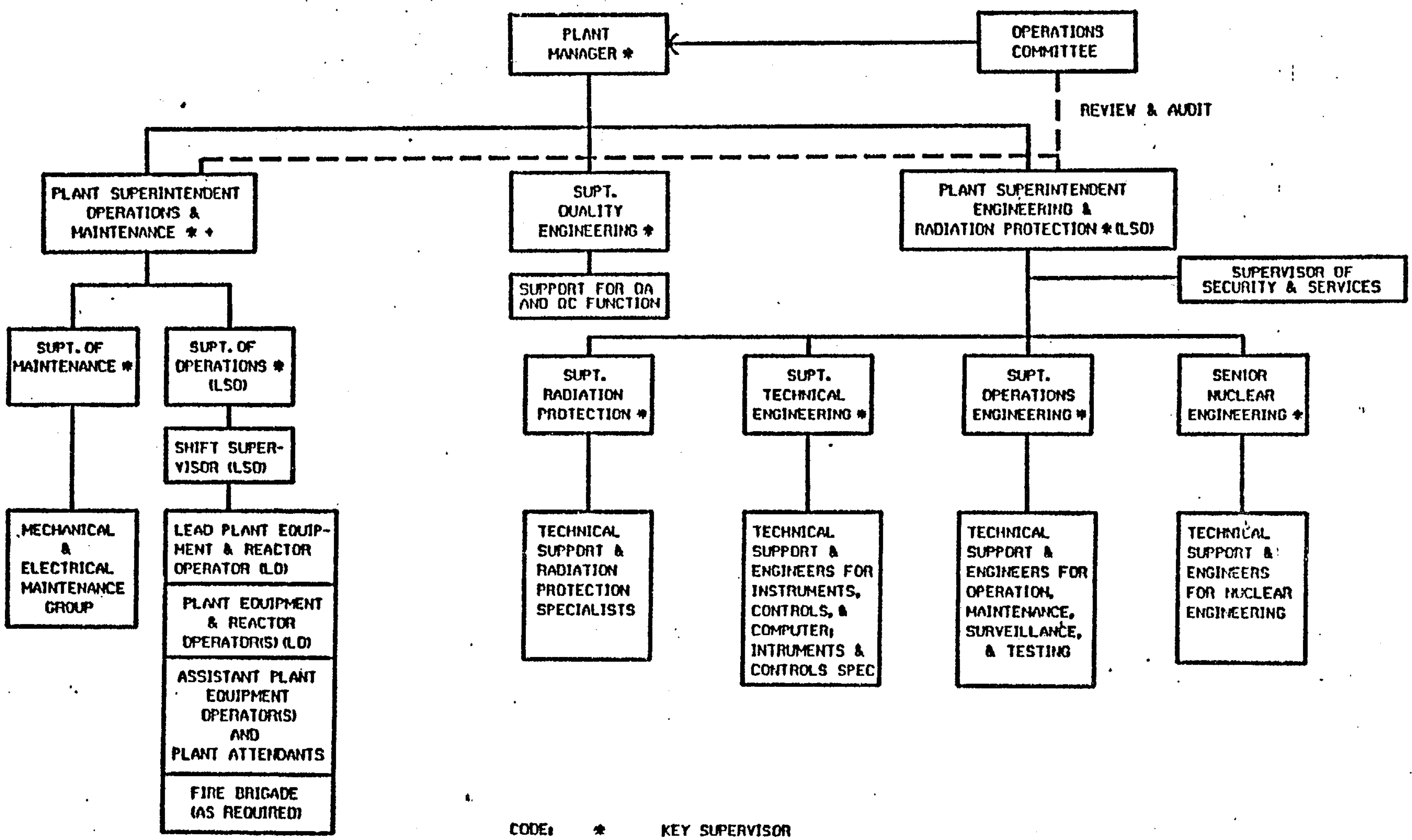
*Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.



* HAS THE RESPONSIBILITY FOR THE FIRE PROTECTION PROGRAM

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



CODE:
- * KEY SUPERVISOR
- LO LICENSED OPERATOR
- LSO LICENSED SENIOR OPERATOR
- + HAS RESPONSIBILITY FOR IMPLEMENTATION OF THE FIRE PROTECTION PROGRAM

FIGURE TS.6.1-2 PRAIRIE ISLAND NUCLEAR GENERATING PLANT FUNCTIONAL ORGANIZATION FOR ON-SITE GROUP

Unit 1 - Amendment No. 39, 49,

Unit 2 - Amendment No. 32, 43, 55

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. <u>Safety Audit Committee (SAC)</u>

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 Director of Nuclear Generation. Other SAC members shall be appointed by the Director of 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 events which are required by regulation or technical specifications (Appendix A) to be reported to NRC in writing within 24 hours.

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 Director of Nuclear Generation, by the SAC at a scheduled meeting, and by members of management having responsibility in the areas audited.

7. Authority

The SAC shall be advisory to the Director of Nuclear Generation.

8. Records

Minutes shall be prepared and retained for all scheduled meetings of the Safety Audit Committtee. The minutes shall be distributed within one month of the meeting to the Director of Nuclear Generation, the General Manager Nuclear Plants, each member of the SAC and others designated by the Chairman. There shall be a formal approval of the minutes.

9. Procedures

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

a. Subjects within the purview of the group.

b. Responsibility and authority of the group.

c. Mechanisms for convening meetings.

d. Provisions for use of specialists or subgroups.

e. Authority to obtain access to the nuclear power plant operating record files and operating personnel when assigned audit functions.

f. Requirements for distribution of reports and minutes prepared by the group to others in the NSP organization.

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 Director of Nuclear Generation and to the Chairman of the Safety Audit Committee.

6.5 PLANT OPERATING PROCEDURES

Detailed written procedures, including the applicable checkoff lists and instructions, covering areas listed below shall be prepared and followed. These procedures and changes thereto, except as specified in TS 6.5.D., shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager.

A. Plant Operations

1. Integrated and system procedures for normal startup, operation and shutdown of the reactor and all systems and components involving nuclear safety of the facility.

2. Fuel handling operations

3. Actions to be taken to correct specific and foreseen potential or actual malfunction of systems or components including responses to alarms, primary system leaks and abnormal reactivity changes and including follow-up actions required after plant protective system actions have initiated.

4. Surveillance and testing requirements that could have an effect on nuclear safety.

5. Implementing procedures of the security plan.

6. Implementing procedures of the Facility Emergency Plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity.

7. Implementing procedures of emergency plans for coping with earthquakes and floods. The flood emergency plan shall require plant shutdown for water levels at the site higher than 692 feet above MSL.

8. Implementing procedures of the fire protection program.

9. Implementing procedures for the Process Control Program and Offsite Dose Calculation Manual including quality control measures.

Drills on the procedures specified in A.3. above, shall be conducted as a part of the retraining program.

B. Radiological

Radiation control procedures shall be maintained and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10CFR20. This radiation protection program shall be organized to meet the requirements of 10CFR20.

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

B. Reportable occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date. Unless explicitly stated, the requirements of this section do not apply to the fire protection systems and measures contained in Sections 3.14/4.16, the radiological effluent limitations and measures in Sections 3.9/4.17, or the radiological environmental monitoring program in Section 4.10. Fire protection reporting requirements have been separately specified in those sections.

1. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Administrator of the appropriate Regional NRC Office or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

   (a) Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

   Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items B.1(e), B.1(f), or B.2(a) below.

   (b) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

   Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item B.2(b) below.

   (c) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

   Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. 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 September 14, 1982, Northern States Power Company (NSP or 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 propose changes in the Technical Specifications (TS) in the following areas.

1. Table TS.4.2-1, Inservice Inspection Requirements
2. TS 4.12.B.4 (new), Steam Generator Tube Surveillance
3. Table TS.4.12-1, Steam Generator Tube Surveillance
4. Typographical corrections to TS pages TS 3.1-3, TS 4.5-2, TS 5.6-2 and TS 6.1-1
5. TS 6.0, Administrative Controls
6. TS 6.2.B.4.(b), Updated Safety Analysis Report
7. TS 6.5.A, Plant Operating Procedures
8. TS 6.7.A, Reporting Requirements
9. TS 6.7.B, Reporting Requirements
10. TS 3.3.A.1.(b) and 3.3.A.2.(e), Accumulator Isolation Valve Requirements
11. TS 3.3.D.2.a.(2), Operability of Diesel-driven Cooling Water Pumps

12. TS 3.4.A, Steam Exclusion System
13. TS 3.4.A.2, Auxiliary Feedwater System
14. Table TS.4.1-1, Instrumentation Surveillance

Discussion and Evaluation

1. Table TS.4.2-1, Inservice Inspection Requirements

The licensee has proposed a change to the TS that permits the use of either magnetic particle (M.T.) or liquid penetration (P.T.) surface examination methods for inspecting surfaces of the pump flywheels. M.T. examination, which would be added to the TS by this change, will permit the licensee an alternative to the P.T. method where accessibility in performing the examination appears to be a problem. We find that the M.T. examination method is equivalent to the P.T. method in determining the presence of surface cracks or discontinuities of ferrous materials and both methods of examination meet the requirements of the ASME Section XI Code (IWA2220). The use of the M.T. examination method will in no way reduce the level of safety and therefore we find this change acceptable.

2. TS 4.1.2.B.4 (new), Steam Generator Tube Surveillance

The licensee has proposed to change the method of selecting steam generator tubes for inspection when the inspection results are classified as category C-3. Specifically, the licensee proposes to include the following conditions when category C-3 inspection is required:

- When the sample inspection results in a C-3 category, all tubes will be inspected, including all tubes under the template plugs and eddy current positioning fixture, with the following exceptions:

(a) The defects in the tubes inspected are at specific locations on the tubes. For example, if all defects are located at anti-vibration bars, then only those tubes that come into contact with anti-vibration bars need be inspected.

(b) The defects in the tubes inspected are related to a defined problem. For example, if all defects are located in the Row 1 or 2 short radius U-bend, then only the Row 1 and 2 tubes need be inspected.

The results of the first sample inspection are used to categorize results into C-1, C-2 or C-3 which forms the basis for subsequent sample inspections. The proposed change would influence the steam generator tube selection in category C-3 for subsequent sample inspections and thus affect the inspection results. Whether the affected results will be more or less conservative than the results obtained by the existing sampling methods cannot be established from the licensee's submittal. In addition, the licensee is proposing that the tube selection for subsequent inspections be concentrated in locations where the cause of the defects can be defined. However, the basis for such definitions has not been established. Based on the above evaluation and our review of the licensee's submittal, we find the proposed TS change is unacceptable.

3. Table TS.4.12-1, Steam Generator Surveillance

The existing TS defines S as follows:

$S = \frac{6\%}{n}$ where n is the number of steam generators inspected during an inspection period and S is the total number of tubes to be inspected.

The licensee has proposed to define "S" in an equivalent form, taking into consideration that each unit contains two steam generators. Thus the licensee proposes to define S as follows:

S = 3%, when two steam generators are inspected during that outage;

S = 6%, when one steam generator is inspected during that outage.

The proposed change will in no way reduce the minimum number of steam generator tubes per steam generator that the licensee is required to inspect during the refueling outages. Therefore, the intent of the TS is not compromised by this proposed change. We agree with the licensee that the proposed change minimizes any potential confusion in interpreting the TS. On this basis we have concluded that the proposed change to the TS is acceptable.

4. Typographical correction to TS pages TS 3.1-3, TS 4.5-2, TS 5.6-2 and TS 6.1-1

Based on the licensee's review of the TS, the following typographical errors are proposed to be corrected.

a. Remove the redundant reference on page TS.3.1-3 as noted in Exhibit B. Reference on page TS.3.1-3A.

b. Correct typgraphical error in TS.4.5.B.1.a by changing the word "heat" to "head".

c. Remove the redundant reference "(1) FSAR Section 9" which appears on page TS 5.6-2.

d. Correct typographical error in TS 6.1.A.5 by changing "of Senior Reactor Operator" to "or Senior Reactor Operator".

We have reviewed these typographical corrections and find that the proposed corrections do not change any of the TS requirements nor the intent of the statement that are affected by these proposed corrections. Since these changes serve only to correct errors as described above, they do not involve significant new safety information of a type not considered by a previous Commission safety review of the facility. We therefore find these proposed changes to correct the typographical errors acceptable.

5. TS 6.0, Administrative Control

The licensee has proposed to update the NSP Corporate Organization Chart Figure TS 6.1-1 to reflect the recent organization changes. Other changes include position title changes on pages TS 6.2-1, TS 6.2-3 and TS 6.2-5 to reflect title changes of various positions. The licensee requested to replace Figure TS 6.1-2 with a redrawn figure.

Our review of these proposed changes indicates that the responsiblilities and resources of the revised organization are essentially unchanged. In addition the reporting functions of the Safety Audit Committee and the Directors for Quality Assurance, Maintenance and Nuclear Generation have not changed such that the level of plant safety is reduced, there is not a significant increase in the probability or consequences of an accident and there is not a significant decrease in safety margin. On this basis we conclude that the proposed organization provides an adequate organization arrangement to manage and support the operational status of the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 and therefore the proposed changes are acceptable.

6. TS 6.2.B.4.(b), Updated Safety Analysis Report

The licensee has proposed to revise the title of Final Safety Analysis Report (FSAR) as it appears in TS 6.2.B.4.(b) to the Updated Safety Analysis Report (USAR). The USAR is to be used as the reference document for determining modifications that will be required by Operation Committee Review. The USAR is a new document updating the information in the FSAR pursuant to 10 CFR 50.71(e). This change permits the licensee to use USAR which is an administrative change that does not involve a safety issue. On this basis we find this change acceptable.

7. TS 6.5.A, Plant Operation

The licensee has proposed to delete the semiannual drills on the emergency plan procedures, including checks of communications with the offsite support groups. Specifically, the licensee proposes to delete the following:

> "Drills on the procedures specified in A.6 above shall be conducted at least semiannually including a check of communication with offsite support groups."

A.6 specified "implementing procedures of the emergency plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity."

We agree with the licensee that Emergency Plan drill requirements are covered in detail in 10 CFR 50 Appendix E Part IV paragraphs E and F. These parts of the Code of Federal Regulations require licensees to perform the Emergency Plan Drills including a check of the communication system annually. The existing requirement in TS is inconsistent and confusing when compared with 10 CFR 50 Appendix E. In addition, we find that there is no basis for conducting these exercises more often than annually. We therefore conclude that deleting the drill requirement statement in the TS is acceptable.

8. TS 6.7.A, Reporting Requirement

By our letter dated March 2, 1982 we accepted the licensee's commitment to report relief valve and safety valve challenges. We also requested the licensee to formalize the reporting requirements of these valves. The licensee, in response to our request has proposed this TS change which will require the licensee to submit an annual report on the safety and relief valve failures and challenges prior to March 1 of each year. We find that the licensee has fulfiled our request in formalizing this reporting requirement and therefore this change is acceptable.

9. TS 6.7.B, Reporting Requirement

The licensee proposed a change to clarify the reporting requirement related to fire protection events. The fire protection system reporting requirements are addressed separately in TS Sections 3.14 and 4.16. The proposed clarification merely states that the established reporting procedures for reportable occurrences in TS Section 6.7.B does not apply for fire protection systems. Similar clarification wording has been found acceptable by the staff for the Monticello plant pursuant to the amendment issued by letter dated June 30, 1981. We have reviewed the proposed clarification and find that the clarification separates the reporting requirements for fire protection from other reporting requirements called for under the administrative section of the TS which is also the purpose of identifying the reporting requirements in TS Section 3.14 and 4.16 for fire protection. On this basis we find the clarification for the reporting requirement related to fire protection events to be acceptable.

10. <u>TS 3.3.A.1.(b) and 3.3.A.2.(e), Accumulator Isolation Valve Requirements</u>

The licensee has proposed to reword TS 3.3.A.1.(b) and 3.3.A.2.(e) related to the operability of the reactor coolant system accumulators in order to avoid any possible misinterpretation of the TS. The existing TS reads as follows:

TS 3.3.A.1.(b), "Each reactor coolant system accumulator shall be operable except that each may be isolated below a pressurizer pressure of 1000 psig, and

TS 3.3.A.2.(e), "One accumulator may be inoperable for up to one hour".

The proposed rewording of these two TS would read as follows:

TS 3.3.A.1.(b), "Each reactor coolant system accumulator shall be operable when the reactor system pressure is greater than 1000 psig" and

TS 3.3.A.2.(e), "One accumulator may be inoperable for up to one hour whenever pressurizer pressure is greater than 1000 psig.

We agree with the licensee that both the existing and the proposed TS require that both accumulators must be operable when the pressurizer pressure is above 1000 psig and that one of the two accumulators can be isolated for up to one hour. The proposed rewording does not change the TS requirements but does eliminate the confusing wording that exists in the current TS. On this basis we find the proposed rewording on accumulator operability acceptable.

11. TS 3.3.D.2.a.(2), Operability of Diesel Driven Cooling Water Pump

The licensee has proposed to show the complete title of the pump reference in TS 3.3.D.2.a.(2) in order to eliminate any possible confusion. The words to be added by the proposed change are "the operable diesel driven cooling water." The proposed change is editorial in nature and does not alter in any way the intent of the TS. We find this change acceptable.

12. TS 3.4.A, Steam Exclusion System

Discussion and Background

The steam exclusion system isolates the ventilation ducts that penetrate rooms containing equipment required to bring the reactor to safe shutdown from plant areas outside containment containing high energy lines. The redundant dampers that exist in these ducts prevent high temperature steam from entering the rooms containing safeguard equipment in the unlikely event that a high energy line break occurs outside containment. The existing TS requires the licensee to have both isolation dampers in each ventilation duct operable or have one damper closed, if one of the redundant dampers is inoperable. This TS has been interpretted by the licensee to mean that during the testing period, one damper of the redundant dampers shall be closed. As a result of this interpretation, the room environmental controls are disrupted due to the lack of circulating air during the testing period. In addition the dampers are exercised each time the temperature sensors are actuated by the test signal, resulting in the dampers being opened and closed an excessive number of times, causing unnecessary damper wear. Such problems had not been forseen at the time the TS were issued. The licensee, in order to resolve this issue, has proposed the following:

a. Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside of containment shall be operable.

b. If one of the two redundant dampers is removed from service, or found inoperable, the operable damper shall be tested daily. If after 48 hours, the inoperative damper is not returned to service, one of the two dampers shall be closed.

c. The actuation logic for one train of steam exclusion may be out of service for 48 hours provided the other train is tested to demonstrate operability prior to initiating repair of the inoperable channel and every 24 hours thereafter.

Evaluation and Conclusion

We agree with the licensee that some time should elapse before the redundant damper is closed to permit testing and maintenance of the dampers. However, allowing 48 hours before the redundant damper is closed has been judged excessive since, if after 48 hours the damper cannot be closed, the licensee has an additional 48 hours before the reactor is brought to cold shutdown, thus permitting a total of 96 hours to elapse from the time of a known failure of a redundant component to the time the plant is brought to cold shutdown. In addition other backup safety components are allowed to be inoperable for periods up to 72 hours (e.g., turbine driven auxiliary feedwater pump). On this basis we requested that the proposed TS be modified to read as follows:

a. Both isolation dampers in each ventilation duct that penetrates rooms containing equipment required for a high energy line rupture outside containment shall be operable.

b. If one of the two redundant dampers is removed from service or for testing and maintenance purposes or found inoperable, the operable redundant damper may remain open for a period not to exceed 24 hours. If after 24 hours, the inoperative damper is not returned to service, one of the two dampers shall be closed.

c. The actuation logic for one train of steam exclusion may be out of service for 24 hours provided the other train is tested and found operable prior to initiating repair of the inoperable channel.

The requested modifications will provide adequate time to perform the necessary testing and maintenance of the steam exclusion system without interrupting the environmental controls of the plant. In addition this modification will also permit the licensee to reduce the number of unnecessary damper closings when the steam exclusion system is tested. Our proposed modifications to the licensee's TS change request was discussed with and agreed to by the licensee.

Operating experience (i.e., from plant startup to present) has shown that after 2810 damper tests in the steam exclusion system there have been only six occasions where dampers were found inoperable. Therefore, the proposed change as modified does not in any way reduce the level of safety. On this basis we find the licensee's proposed TS change related to the steam exclusion system dampers as modified is acceptable.

13. TS 3.4..2, Auxiliary Feedwater System

The licensee reevaluated the auxiliary feedwater system based on the criteria in NUREG-0737 Items II.E.1.1 and II.E.1.2. Based on our review of the licensee's submittals on this matter, by letter dated March 22, 1982, we issued our safety evaluation in which we requested the licensee to commit to submitting a TS change request requiring assurance that cross tie valves in cross connects between the condensate tanks be locked in an open position. This proposed TS is in response to this commitment. The proposed TS requires the cross tie valves to be blocked and tagged open. Any change in position of the valves will be under direct administrative control. We find that the licensee has fulfilled his commitment as discussed in our safety evaluation (March 22, 1982) and therefore the proposed change is acceptable.

14. Table TS.4.1-1, Inservice Inspection Requirement

As part of the licensee's reevaluation of the auxiliary feedwater system under the criteria of NUREG-0737 Items II.E.1.1 and II.E.1.2, the licensee committed to install suction and discharge pressure switches. By letter dated January 7, 1982 the licensee further committed to submit a TS change request governing the surveillance and operability of these pressure switches. This TS change request is the licensee's fulfillment of this commitment. The pressure switch serves as protection for the auxiliary feedwater pumps from damage due to loss of suction pressure and pump runout conditions. Our technical basis requiring the installation of these pressure switches is discussed in our safety evaluation issued to the licensee by letter dated March 22, 1982 as related to meeting the requirement of GL-4 (long term recommendations). The proposed TS change will require the licensee to check the calibration and perform a functional test during each refueling outage. Based on the above evaluation we find the proposed change acceptable.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: February 2, 1983

Principal Contributor:
D. C. DiIanni

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 61 and 55 to Facility Operating License Nos. DPR-42 and DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Technical Specifications (TS) concerned with Steam Generator Tube Surveillance, Steam Exclusion System, Auxiliary Feedwater System reporting requirements, Administrative Controls, and correct typographical errors.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated September 14, 1982, (2) Amendment Nos. 61 and 55 to License Nos. DPR-42 and DPR-60, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Environmental Conservation Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 2nd day of February, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing