EXHIBIT B

License Amendment Request Dated September 14, 1982

Exhibit B, attached, consists of the following revised pages for the Appendix A Technical Specifications which incorporate the proposed changes.

PAGES

TS 3.1-3
TS 3.3-1
TS 3.3-2
TS 3.3-5A
TS 3.4-2
TS 3.4-3
Table TS.4.1-1 (Pg. 2 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. 5 of 5)
Table TS.4.2-1
TS 4.5-2
TS 4.12-2
TS 4.12-2A (new page)
Table TS 4.12-1
TS 5.6-2
TS 6.1-1
Figure TS 6.1-1 & TS.6.1-2
TS 6.2-1
TS 6.2-3
TS 6.2-5
TS 6.5-1
TS 6.7-2

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 genertor tube imperfections having a depth ≥ 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 Specificiation 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. Containment Cooling Systems

1. A reactor shall not be made or maintained critical nor shall it be heated above 200°F unless the following conditions are satisifed 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 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.

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 2a or 2b, 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.

Reference

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

TABLE TS.4.1-1
(Page 2 of 5)

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 Level	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.	Radiation Monitoring	*D	R	M	NA	Includes all channels used for leak detection per Spec. 3.1 D. and effluent release monitoring per Spec. 3.9 and 5.5.
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

TABLE TS.4.1-1
(Page 4 of 5)

Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
27. Turbine Overspeed Protection Trip Channel	NA	R	M	NA	
28. Deleted					
29. Deleted					
30. Environmental Monitors	M	NA	NA	NA	Includes those used per Spec. 4.10
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 Instruments	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 Apendix 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	(1)
41.	Auxiliary Feedwater Pump Discharge Pressure	NA	R	R	NA	(1)

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

(1) Installed and operational for Unit 2. Will be installed for Unit 1 pumps during 1982 refueling outage and will then be operable.

TABLE TS.4.2-1
REV

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 examinaton 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 & PV 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.

2. The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:

   (a) All nonplugged tubes that previously had detectable wall penetrations (> 20%).

   (b) Tubes in those areas where experience has indicated potential problems.

   (c) A tube inspection (pursuant to Specification 4.12.D.1.(h)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

3. The tubes selected as the second and third samples (if required by Table TS.4.12-1) during each inservice inspection may be subjected to a partial tube inspection provided:

   (a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

   (b) The inspections include those portions of the tubes where imperfections were previously found.

4. 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 each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

C-2 One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.

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-handle 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 responsibility to other qualified supervisory personnel.

B. The Northern States Power corporation 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.

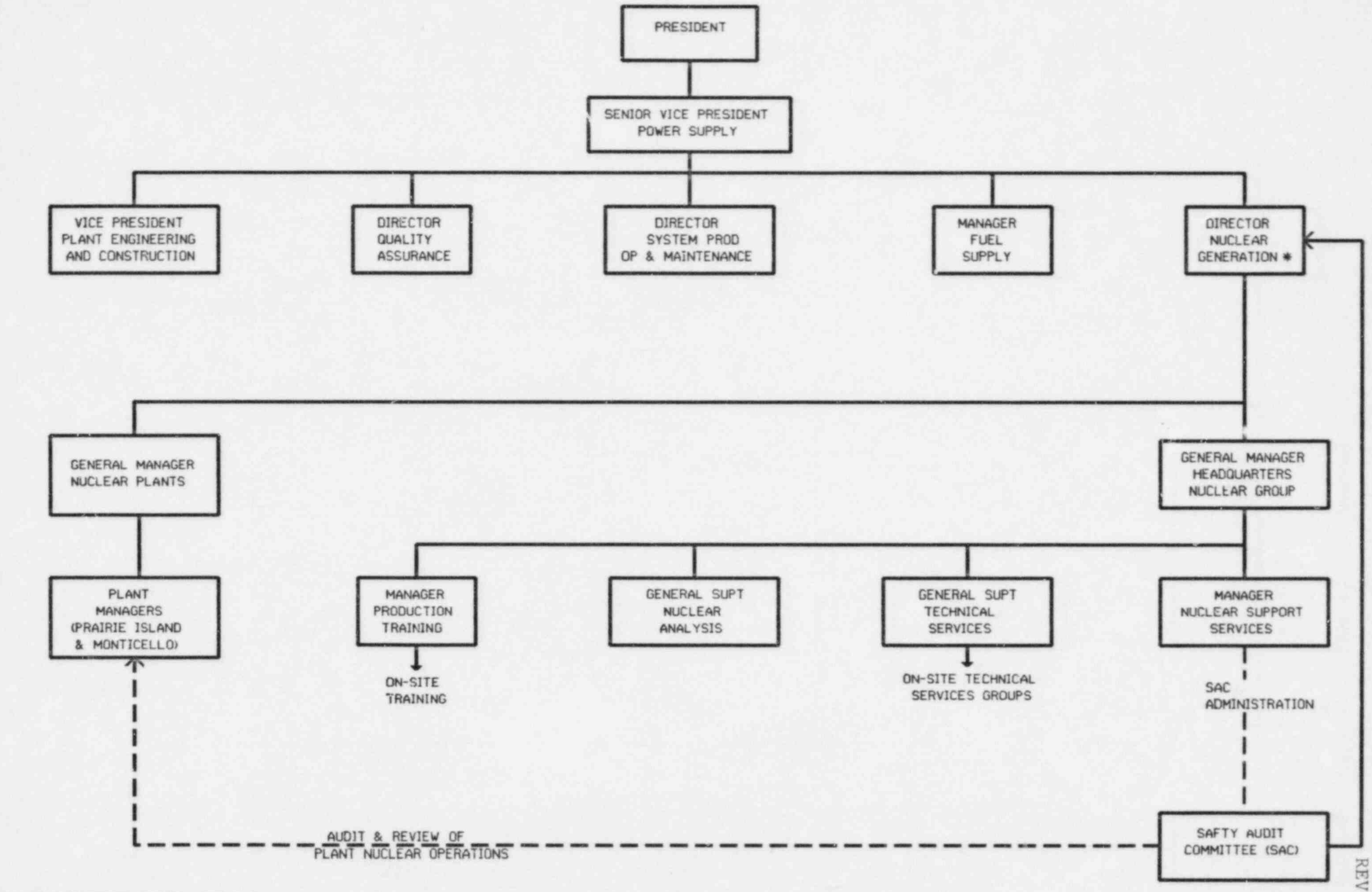
4. An individual qualified in radiation protection procedures shall be on site when fuel is in a reactor.

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.

---

*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

CAD ORTS611

FIGURE TS.6.1-1
REV

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

LAD 0RT5612

6.2 Review and Audit

Organizational units for the ew 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 a 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 plants.

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.

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

   c. 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 Committee. 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.

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 individuals 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.

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.

2. Occupational Exposure Report.[1/] An annual report of occupational exposure covering the previous calendar year shall be submitted prior to March 1 of each year.

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

3. Monthly Operating Report A monthly report of operating statistics and shutdown experience covering the previous month shall be submitted by the 15th of the following month to the Office of Management Information and Program Control, U S Nuclear Regulatory Commission, Washington, D C 20555.

4. Steam Generator Tube Inservice Inspection. The results of steam generator tube inservice inspections shall be reported within 90 days of January 1 for all inspections completed during the previous calendar year. These reports shall include; (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged.

5. 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 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. The requirements of this section do not apply to the fire protection systems and measures contained in Sections 3.14/4.16 of the Technical Specifications. Fire protection reporting requirements have been separately specified in those sections.

1/ This report supplements the requirements of 10CFR20, Section 20.407. If 10CFR20, Section 20.407 is revised to include such information, this Specification is unnecessary.

Next page is 6.7-4