EXHIBIT B

License Amendment Request dated December 12, 1980

Docket No. 50-263
License No. DPR-22

Exhibit B consists of revised pages of Appendix A Technical Specifications as listed below:

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121B
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121D (new page)
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*Previously submitted with License Amendment Request dated May 15, 1980.

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LICENSE AMENDMENT REQUEST

TABLE 3.2.1 - Continued

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions
b. High Drywell Pressure (5)	≤ 2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. Low Reactor Water Level	≥ 10'6" above the top of the active fuel	2	2	E
b. High Drywell Pressure	≤ 2 psig	2	2	E
4. HPCI Steam Lines				
a. HPCI High Steam Flow	≤150,000 lb/hr with ≤60 second time delay	2(4)	2	F
b. HPCI High Steam Flow	≤300,000 lb/hr	2(4)	2	F
c. HPCI Steam Line Area High Temp.	≤200°F	16(4)	16	F
5. RCIC Steam Lines				
a. RCIC High Steam Flow	≤45,000 lb/hr	2(4)	2	G
b. RCIC Steam Line Area	≤200°F	16(4)	16	G

Bases Continued:

3.2 This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, and RWCU system valves.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow,

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVL	ACCESSIBLE -A INACCESSIBLE-I
PS1-H2	MAIN STEAM	DRYWELL	953	071			I
PS1-H3	MAIN STEAM	DRYWELL	950	148			I
PS2-H2	MAIN STEAM	DRYWELL	950	120			I
PS3-H2	MAIN STEAM	DRYWELL	950	240			I
PS4-H3	MAIN STEAM	DRYWELL	950	212			I
RV24-H3	SAFETY-RELIEF	DRYWELL	950	110			I
RV24-H4	SAFETY-RELIEF	DRYWELL	935	100			I
RV24-H4A	SAFETY-RELIEF	DRYWELL	935	100			I
RV24-H5	SAFETY-RELIEF	DRYWELL	935	110			I
RV24-N1	SAFETY-RELIEF	DRYWELL	953	090		X	I
RV24A-H4A	SAFETY-RELIEF	DRYWELL	947	048		X	I
RV24A-H7	SAFETY-RELIEF	DRYWELL	953	115			I
RV24A-H8	SAFETY-RELIEF	DRYWELL	939	032			I
RV24A-N1	SAFETY-RELIEF	DRYWELL	956	086		X	I
RV25-H1	SAFETY-RELIEF	DRYWELL	953	180			I
RV25-H1A	SAFETY-RELIEF	DRYWELL	953	180	X		I
RV25-H2	SAFETY-RELIEF	DRYWELL	948	190		X	I
RV25-H2A	SAFETY-RELIEF	DRYWELL	948	190		X	I
RV25-H3	SAFETY-RELIEF	DRYWELL	934	180	X		I
RV25A-H2	SAFETY-RELIEF	DRYWELL	945	120	X	X	I
RV25A-H2A	SAFETY-RELIEF	DRYWELL	945	120	X	X	I
RV25A-H7	SAFETY-RELIEF	DRYWELL	953	135			I
RV26-H1	SAFETY-RELIEF	DRYWELL	953	200	X		I
RV26-H1A	SAFETY-RELIEF	DRYWELL	953	200			I
RV26-H2	SAFETY-RELIEF	DRYWELL	947	200		X	I
RV26-H2A	SAFETY-RELIEF	DRYWELL	947	200			I
RV26-N1	SAFETY-RELIEF	DRYWELL	956	200	X	X	I
RV26A-H2	SAFETY-RELIEF	DRYWELL	940	250			I
RV26A-H2A	SAFETY-RELIEF	DRYWELL	935	250			I
RV26A-N1	SAFETY-RELIEF	DRYWELL	950	250	X	X	I
RV26A-N2	SAFETY-RELIEF	DRYWELL	951	250	X	X	I
RV27-H1	SAFETY-RELIEF	DRYWELL	950	320			I
RV27-H1A	SAFETY-RELIEF	DRYWELL	950	230			I
RV27-H5	SAFETY-RELIEF	DRYWELL	945	270			I
RV27-H6	SAFETY-RELIEF	DRYWELL	945	270			I
RV27-N1	SAFETY-RELIEF	DRYWELL	956	270	X	X	I
RV27A-H2A	SAFETY-RELIEF	DRYWELL	953	290			I
RV27A-H5	SAFETY-RELIEF	DRYWELL	953	290			I
RV27A-H9	SAFETY-RELIEF	DRYWELL	938	290			I
RV27A-N1	SAFETY-RELIEF	DRYWELL	956	270	X	X	I
SS-1	MAIN STEAM	DRYWELL	953	279	X		I
SS-1AR	RECIRCULATION	DRYWELL	922	315	X	X	I
SS-1BR	RECIRCULATION	DRYWELL	922	135	X	X	I
SS-11	FEEDWATER	DRYWELL	952	302			I
SS-12	FEEDWATER	DRYWELL	952	058			I
SS-13	FEEDWATER	DRYWELL	952	258			I
SS-14	FEEDWATER	DRYWELL	952	096			I

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
SS-17A	RHR	DRYWELL	964	072	X		I
SS-17B	RHR	DRYWELL	964	072	X		I
SS-18A	RHR	DRYWELL	964	288			I
SS-18B	RHR	DRYWELL	964	288			I
SS-19	RHR	DRYWELL	964	341			I
SS-2	MAIN STEAM	DRYWELL	953	061	X		I
SS-2AR	RECIRCULATION	DRYWELL	927	302	X	X	I
SS-2BR	RECIRCULATION	DRYWELL	927	122		X	I
SS-2C	RHR	DRYWELL	964	019	X		I
SS-3	MAIN STEAM	DRYWELL	950	212			I
SS-3AR	RECIRCULATION	DRYWELL	927	328		X	I
SS-3BR	RECIRCULATION	DRYWELL	927	148		X	I
SS-4	MAIN STEAM	DRYWELL	950	148			I
SS-4AR(A)	RECIRCULATION	DRYWELL	934	302			I
SS-4AR(B)	RECIRCULATION	DRYWELL	934	323			I
SS-4BR(A)	RECIRCULATION	DRYWELL	934	120			I
SS-4BR(B)	RECIRCULATION	DRYWELL	934	149			I
SS-40	HPCI	MAIN STEAM CHASE					I
SS-5AR	RECIRCULATION	DRYWELL	941	315		X	I
SS-5BR	RECIRCULATION	DRYWELL	941	135		X	I
SS-6AR	RECIRCULATION	DRYWELL	953	261	X		I
SS-6BR	RECIRCULATION	DRYWELL	953	099	X		I
SS-7	MAIN STEAM	DRYWELL	953	240	X		I
SS-7AR	RECIRCULATION	DRYWELL	953	323			I
SS-7BR	RECIRCULATION	DRYWELL	953	032			I
SS-8	MAIN STEAM	DRYWELL	953	120	X		I
SS-8AR	RECIRCULATION	DRYWELL	927	270		X	I
SS-8BR	RECIRCULATION	DRYWELL	927	090		X	I
SS-21	RHR	TORUS FL LV - S WALL					A
SS-22	RHR	TORUS FL LV - S WALL					A
SS-23	RHR	B RHR ROOM FL LV					A
SS-24	RHR	A RHR ROOM FL LV					A
SS-25	RHR	TORUS CATWK-SE WALL					A
SS-26	CORE SPRAY	B RHR ROOM FL LVL					A
SS-27	CORE SPRAY	B RHR ROOM FL LVL					A
SS-28A	CORE SPRAY	A RHR ROOM FL LVL					A
SS-28B	CORE SPRAY	A RHR ROOM FL LVL					A
SS-29	RHR	OVER N2 ANALYZER	954			X	A
SS-30	RHR	OVER N2 ANALYZER	954			X	A
SS-31	RHR	TORUS CATWK					A
SS-32A	RHR	A RHR ROOM - BY HX	916			X	A
SS-32B	RHR	A RHR ROOM - BY HX	916			X	A
SS-33	RHR	ABOVE TORUS					A
SS-34	RHR	ABOVE TORUS					A
SS-35	HPCI	HPCI ROOM - N WALL	912			X	A
SS-36A	HPCI	HPCI ROOM - FL LVL					A
SS-36B	HPCI	HPCI ROOM - FL LVL					A

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	HIGH RADIATION AREA	DIFFICULT TO REMOVE	ACCESSIBLE -A INACCESSIBLE-I
SS-37	HPCI	HPCI ROOM - W WALL	905			X	A
SS-38A	RCIC	RCIC ROOM - W WALL	906			X	A
SS-38B	RCIC	RCIC ROOM - W WALL	906			X	A
SS-41	CORE SPRAY	ABOVE TORUS CATWK	927				A
SS-42	HPCI	ABOVE TORUS RING HDR	906				A

3.14 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

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

Objective:

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

Specification:

Whenever the reactor is in the startup or run mode, the limiting conditions for operation for accident monitoring instrumentation given in Table 3.14.1 shall be satisfied.

4.14 ACCIDENT MONITORING INSTRUMENTATION

Applicability:

Applies to the surveillance requirements for accident monitoring instrumentation.

Objective:

To specify the type and frequency of surveillance to be applied to accident monitoring instrumentation.

Specification:

The accident monitoring instrumentation shall be functionally tested and calibrated in accordance with Table 4.14.1.

Table 3.14.1

Instrumentation for Accident Monitoring

Function	Total No. of Instrument Channels	Minimum No. of Operable Channels	Required Conditions*
Reactor Vessel Water Level (Fuel Zone)	2	1	A
Safety/Relief Valve Position (One Channel Pressure Switch and One Channel Thermocouple Position Indication per Valve)	2/Valve	1/Valve	A

*Required conditions:

A. When the number of channels made or found to be inoperable is less than the minimum number of operable channels shown, either restore the inoperable channel(s) to operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

Table 4.14.1
Minimum Test and Calibration Frequency for
Accident Monitoring Instrumentation

Instrument Channel	Test (Note 1)	Calibration (Note 1)	Sensor Check (Note 1)
Reactor Vessel Water Level (Fuel Zone)	-	Once/Operating Cycle	Once/month (Note 3)
Safety/Relief Valve Position (Pressure Switches)	-	Once/Operating Cycle	Note 2
Safety/Relief Valve Position (Thermocouples)	-	Once/Operating Cycle	Note 2

NOTES:

(1) Functional tests, calibrations, and sensor checks are not required when the instruments are not required to be operable. If tests are missed, they shall be performed prior to returning the instruments to an operable status.

(2) Proper instrument response shall be verified during each safety/relief valve actuation.

(3) These instruments are off-scale high during normal plant operation.

Bases:

3.14/4.14 The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Learned Task Force Status Report and Short Term Recommendations".

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 corporate organizational structure relating to the operation of the plant is shown in Figure 6.1.1.

C. The minimum functional organization for operation of the plant shall be as shown in Figure 6.1.2 and:

1. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.1.1.

2. At least one licensed operator shall be in the control room when fuel is in the reactor.

3. At least two licensed operators shall be present in the control room during cold startup, scheduled reactor shutdown, and during recovery from reactor trips.

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

5. All alterations of the reactor core shall be directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6. A fire brigade of at least three members shall be maintained on site at all times. The fire brigade shall not include the four members of the shift organization required for safe shutdown of the reactor or more than one member of the site security force.

D. Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Superintendent Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. The training program shall be under the direction of a designated member of Northern States Power management.

TABLE 6.1.1

MINIMUM SHIFT CREW COMOSITION (Note 1)

CATEGORY	APPLICABLE PLANT CONDITIONS	
	COLD SHUTDOWN OR REFUELING OPERATION	ABOVE COLD SHUTDOWN
No. Licensed Senior Operators (LSO)	1 (Note 2)	1
Total No. Licensed Operators (LSO & LO)	2	3
Total No. Licensed and Unlicensed Operators	3	5
Shift Technical Advisor	0	1

NOTES:

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

2. Does not include the licensed Senior Reactor Operator, or Senior Reactor Operator Limited to Fuel Handling, supervising alterations of the reactor core.

1. a. Paragraph 20.203 "Caution signs, lables, signals and controls." In lieu of the "Control device" or alarm signal required by paragraph 20.203(c)(2), each high radiation area in which the intensity of radiation is 1000 mRem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mRem/hr, except that locked doors shall be provided to prevent unauthorized entry into these areas and the keys to these locked doors shall be maintained under the administrative control of the Plant Manager.

2. A program shall be implemented to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

   a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

   b. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

   A program acceptable to the Commission was described in a letter dated December 31, 1979, from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, "Lessons Learned Implementation".

3. A program shall be implemented which will ensure the capability to accurately determine the airborne iodine concentration in essential plant areas under accident conditions. This program shall include the following:

   a. Training of personnel,

   b. Procedures for mnitoring, and

   c. Provisions for maintenance of sampling and analysis equipment.

   A program acceptable to the Commission was described in a letter dated December 31, 1979, from L O Mayer, NSP, to Director of Nuclear Reactor Regulation, "Lessons Learned Implementation".