



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. 63
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 October 29, 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) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 63, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION


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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 23, 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. 57
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 October 29, 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.

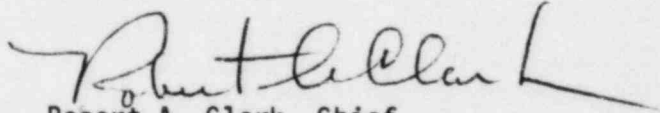
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-60 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 57, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 23, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NOS. 63 AND 57 TO FACILITY OPERATING LICENSE NOS.
DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

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

Remove

TS.i
TS.iii
TS.3.6-2
TS.3.8-1
TS.3.8-4
Table TS.3.12-1 (pg. 1 of 8)
Table TS.3.12-1 (pg. 2 of 8)
Table TS.3.12-1 (pg. 7 of 8)
Table TS.3.12-1 (pg. 8 of 8)
TS.3.15-1

Table TS.3.15-1

Table TS.4.1-1 (pg. 4 of 5)
TS.4.5-2

Insert

TS.i
TS.iii
TS.3.6-2
TS.3.8-1
TS.3.8-4
Table TS.3.12-1 (pg. 1 of 8)
Table TS.3.12-1 (pg. 2 of 8)
Table TS.3.12-1 (pg. 7 of 8)
Table TS.3.12-1 (pg. 8 of 8)
TS.3.15-1
TS.3.15-2
Table TS.3.15-1
Table TS.3.15-2
Table TS.4.1-1 (pg. 4 of 5)
TS.4.5-2

TECHNICAL SPECIFICATIONSTABLE OF CONTENTS

<u>TS SECTION</u>	<u>TITLE</u>	<u>PAGE</u>
1.0	Definitions	TS.1-1
2.0	<u>Safety Limits and Limiting Safety System Settings</u>	
2.1	Safety Limit, Reactor Core	TS.2.1-1
2.2	Safety Limit, Reactor Coolant System Pressure	TS.2.1-1
2.3	Limiting Safety System Settings, Protective Instrumentation	TS.2.2-1
3.0	<u>Limiting Conditions for Operation</u>	TS.2.3-1
3.1	Reactor Coolant System	TS.3.1-1
3.2	Chemical and Volume Control System	TS.3.1-1
3.3	Engineered Safety Features	TS.3.2-1
3.4	Steam and Power Conversion System	TS.3.3-1
3.5	Instrumentation System	TS.3.4-1
3.6	Containment System	TS.3.5-1
3.7	Auxiliary Electrical Systems	TS.3.6-1
3.8	Refueling and Fuel Handling	TS.3.7-1
3.9	Radioactive Effluents	TS.3.8-1
3.10	Control Rod and Power Distribution Limits	TS.3.9-1
3.11	Core Surveillance Instrumentation	TS.3.10-1
3.12	Snubbers	TS.3.11-1
3.13	Control Room Air Treatment System	TS.3.12-1
3.14	Fire Detection and Protection Systems	TS.3.13-1
3.15	Event Monitoring Instrumentation	TS.3.14-1
4.0	<u>Surveillance Requirements</u>	TS.3.15-1
4.1	Operational Safety Review	TS.4.1-1
4.2	Inservice Inspection and Testing of Pumps and Valves Requirements	TS.4.1-1
4.3	Primary Coolant System Pressure Isolation Valves	TS.4.2-1
4.4	Containment System Tests	TS.4.3-1
4.5	Engineered Safety Features	TS.4.4-1
4.6	Periodic Testing of Emergency Power System	TS.4.5-1
4.7	Main Steam Stop Valves	TS.4.6-1
4.8	Steam and Power Conversion Systems	TS.4.7-1
4.9	Reactivity Anomalies	TS.4.8-1
4.10	Radiation Environmental Monitoring Program	TS.4.9-1
4.11	Radioactive Source Leakage Test	TS.4.10-1
4.12	Steam Generator Tube Surveillance	TS.4.11-1
4.13	Snubbers	TS.4.12-1
4.14	Control Room Air Treatment System Tests	TS.4.13-1
4.15	Spent Fuel Pool Special Ventilation System	TS.4.14-1
4.16	Fire Detection and Protection Systems	TS.4.15-1
4.17	Radioactive Effluents Surveillance	TS.4.16-1
		TS.4.17-1

Unit 1 - Amendment No. 50, 59, 63
 Unit 2 - Amendment No. 44, 53, 57

TECHNICAL SPECIFICATIONSLIST OF TABLES

<u>TS TABLE</u>	<u>TITLE</u>
3.1-1	Unit 1 Reactor Vessel Toughness Data (Unirradiated)
3.1-2	Unit 2 Reactor Vessel Toughness Data (Unirradiated)
3.5-1	Engineered Safety Features Initiation Instrument Limiting Set Points
3.5-2	Instrument Operating Conditions for Reactor Trip
3.5-3	Instrument Operating Conditions for Emergency Cooling System
3.5-4	Instrument Operating Conditions for Isolation Functions
3.5-5	Instrument Operating Conditions for Ventilation Systems
3.5-6	Instrument Operating Conditions for Auxiliary Electrical System
3.9-1	Radioactive Liquid Effluent Monitoring Instrumentation
3.9-2	Radioactive Gaseous Effluent Monitoring Instrumentation
3.12-1	Safety Related Snubbers
3.14-1	Safety Related Fire Detection Instruments
3.15-1	Event Monitoring Instrumentation - Process
3.15-2	Event Monitoring Instrumentation - Radiation
4.1-1	Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels
4.1-2A	Minimum Frequencies for Equipment Tests
4.1-2B	Minimum Frequencies for Sampling Tests
4.2-1	Special Inservice Inspection Requirements
4.4-1	Unit 1 and Unit 2 Penetration Designation for Leakage Tests
4.10-1	Radiation Environmental Monitoring Program (REMPO) Sample Collection and Analysis
4.10-2	REMP - Maximum Values for the Lower Limits of Detection
4.10-3	REMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples
4.12-1	Steam Generator Tube Inspection
4.17-1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements
4.17-2	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements
4.17-3	Radioactive Liquid Waste Sampling and Analysis Program
4.17-4	Radioactive Gaseous Waste Sampling and Analysis Program
5.5-1	Anticipated Annual Release of Radioactive Material in Liquid Effluents from Prairie Island Nuclear Generating Plant (Per Unit)
5.5-2	Anticipated Annual Release of Radioactive Nuclides in Gaseous Effluent From Prairie Island Nuclear Generating Plant (Per Unit)
6.1-1	Minimum Shift Crew Composition

Unit 1 - Amendment No. 50, 59, 63

Unit 2 - Amendment No. 44, 52, 57

4. Positive reactivity changes shall not be made by boron dilution when containment system integrity is not intact unless the boron concentration in the reactor is maintained ≥ 2100 ppm for the initial refueling and ≥ 2000 ppm for subsequent refuelings.
5. The vacuum breaker system shall be considered operable for containment system integrity when both valves in each of two vacuum breakers, including actuating and power circuits, are operable or when one vacuum breaker is daily demonstrated as operable and the other has been inoperable for no more than 7 days under conditions for which containment integrity is required.
6. Automatic containment isolation valves listed in Table TS.4.4-1 shall be considered operable for containment system integrity when all automatic isolation valves, including actuation circuits, for each penetration are operable or the inoperable valve is deactivated in the closed position, or at least one valve in each penetration having an inoperable valve is locked closed.
7.
 - a. The 36-inch containment purge system double gasketed blind flanges shall be installed whenever the reactor is above cold shutdown.
 - b. The 18-inch containment inservice purge system double gasketed blind flanges shall be installed whenever the reactor is above cold shutdown except as noted below.
 - c. The inservice purge system may be operated above cold shutdown when required for safe plant operation if the following conditions are met:
 1. The debris screens are installed on the supply and exhaust ducts in containment.
 2. Both valves shall satisfactorily pass a local leak rate test prior to use.
 3. The two automatic primary containment isolation valves and the automatic shield building ventilation damper in each duct that penetrates containment shall be operable, including instruments and controls associated with them.
 4. The blind flanges (i.e., 42E (53 in Unit 2) and 43A (52 in Unit 2) shall be reinstalled and satisfactorily pass a local leak rate test, each time after the inservice purge system is used.
8. During maintenance, construction and testing activities, containment integrity is considered intact if the auxiliary building special vent zone boundary is opened intermittently, provided such openings are under direct administrative control and can be reduced to less than 10 square feet within 6 minutes following an accident.

3.8 REFUELING AND FUEL HANDLING

Applicability

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

Objectives

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

Specification

- A. During refueling operations the following conditions shall be satisfied:
1. The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, at least one isolation valve shall be operable or locked closed in each line which penetrates the containment and provides a direct path from containment atmosphere to the outside.
 2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool areas shall be monitored continuously.
 3. The core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment, which are in service whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.
 4. During reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 2000 ppm shall be maintained in the reactor coolant system. The required boron concentration shall be verified by chemical analysis daily.
 5. During movement of fuel assemblies or control rods out of the reactor vessel, at least 23 feet of water shall be maintained above the reactor vessel flange. The required water level shall be verified prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.
 6. At least one residual heat removal pump shall be operable and running. The pump may be shut down for up to one hour to facilitate movement of fuel or core components.
 7. If the water level above the top of the reactor vessel flange is less than 20 feet, except for control rod latching and unlatching operations, both residual heat removal loops shall be operable.
 8. If Specification 3.8.A.6 or 3.8.A.7 cannot be satisfied, all fuel handling operations in containment shall be suspended, the containment integrity requirements of Specification 3.8.A.1 shall be satisfied, and no reduction in reactor coolant boron concentration shall be made.

the cask into a carrier, there is a potential drop of 66 feet⁽⁵⁾. The cask will not be loaded onto the carrier for shipment prior to a 3-month storage period. At this time, the radioactivity has decayed so that a release of fission products from all fuel assemblies in the cask would result in off-site doses less than 10 CFR Part 100. It is assumed, for this dose analysis, that 12 assemblies rupture after storage for 90 days. Other assumptions are the same as those used in the dropped fuel assembly accident in the SER, Section 15. The resultant doses at the site boundary are 94 Rems to the thyroid and 1 Rem whole body.

The Spent Fuel Pool Special Ventilation System⁽⁴⁾ is a safeguards system which maintains a negative pressure in the spent fuel enclosure upon detection of high area radiation. The Spent Fuel Pool Normal Ventilation system is automatically isolated and exhaust air is drawn through filter modules containing a roughing filter, particulate filter, and a charcoal filter before discharge to the environment via one of the Shield Building exhaust stacks. Two completely redundant trains are provided. The exhaust fan and filter of each train are shared with the corresponding train of the Containment In-service Purge System. High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers in each SFPSVS filter train. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a HEPA filter leakage of less than 1% through DOP testing and a charcoal adsorber leakage of less than 1% through halogenated hydrocarbon testing. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90% under test conditions which are more severe than accident conditions. The satisfactory completion of these periodic tests combined with the qualification testing conducted on new filters and adsorber provide a high level of assurance that the emergency air treatment systems will perform as predicted in the accident analyses.

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

The water level may be lowered to the top of the RCCA drive shafts for latching and unlatching. The basis for this allowance is (1) the refueling cavity pool has sufficient level to allow time to initiate repairs or emergency procedures to cool the core, (2) during latching and unlatching the level is closely monitored because the activity uses this level as a reference point, (3) the time spent at this level is minimal.

References

- (1) FSAR Section 9.5.2
- (2) FSAR Table 3.2.1-1
- (3) FSAR Section 14.2.1
- (4) FSAR Section 9.6
- (5) FSAR Page 9.5-20a

Unit 1 - Amendment No. 25, 47, 63
 Unit 2 - Amendment No. 19, 41, 57

SAFETY RELATED SNUBBERS

<u>Snubber No.</u>	<u>Location</u>	<u>Elevation</u>	<u>Accessible or Inaccessible (A or I)</u>	<u>Snubbers Especially Difficult to Remove</u>	<u>In High Radiation Area During Shutdown</u>
<u>UNIT I</u>					
AFSH-22	A&B Main and Aux-	773'-4½"	A		
AFSH-36	iliary Steam	745'-7½"	A		
AFSH-39		699'-10½"	A		
AFSH-48		699'-6½"	A		
MSDH-25	A&B	736'-6-7/16"	A		
MSDH-26	A&B	756'-7½"	A		
MSDH-29		756'-7½"	A		
MSDH-30		736'-6-7/16"	A		
MSH-48	A&B	739'-1-11/16"	A		
MSH-62	A&B	735'-6"	A		
MSH-63		756'-0"	A		
MSH-64		743'-0"	A		
MSH-65		748'-0"	A		
MSH-66		753'-0"	A		
MSH-67		743'-0"	A		
MSH-68	A&B	755'-8"	A		
MSH-69	A&B	748'-0"	A		
MSH-101		729'-0"	A		
MSH-102		735'-0"	A		
MSH-103	A&B	737'-0"	A		
<u>UNIT II</u>					
AFSH-2	Main and Aux-	749'-4"	A		
AFSH-19	iliary Stear	745'-7½"	A		
AFSH-20		745'-7½"	A		
AFSH-24		745'-6"	A		
AFSH-29	A&B	721'-1-9/16"	A		
AFSH-33		707'-5"	A		
AFSH-39		696'-6½"	A		
AFSH-40		696'-6½"	A		
AFSH-44		750'-7½"	A		
AFSH-46		750'-7"	A		
MSDH-17		739'-0"	A		
MSDH-18		759'-0"	A		
MSDH-19		739'-0"	A		
MSDH-20		759'-0"	A		

Unit 1 - Amendment No. 1A, A9, B0, 63

Unit 2 - Amendment No. B, AB, AA, 57

SAFETY RELATED SNUBBERS

<u>Snubber No.</u>	<u>Location</u>	<u>Elevation</u>	<u>Accessible or Inaccessible (A or I)</u>	<u>Snubbers Especially Difficult to Remove</u>	<u>In High Radiation Area During Shutdown</u>
<u>UNIT II</u>					
MSH-23 A&B	Main and Aux-	739'-1-3/16"	A		
MSH-54 A&B	iliary Steam	756'-0-1/16"	I		
MSH-75		744'-0"	A		
MSH-76 A&B		748'-0"	A		
MSH-77		748'-0"	A		
MSH-78		743'-0"	A		
MSH-79		753'-0"	A		
MSH-80		755'-0"	A		
MSH-81 A&B		735'-9"	A		
MSH-82 A&B		755'-8"	A		
MSH-83		761'-13/16"	I		
MSH-101		727'-0"	A		
MSH-102		734'-0"	A		
MSH-103A&B		736'-0"	A		
<u>UNIT I</u>					
RHRRH-5	Safety Injection	723'-4 1/2"	I		
RHRRH-41		698'-11"	I		
RHRRH-58		670'-0"	A		
RHRRH-60		670'-0"	A		
RPCH-160		718'-1 1/2"	I		
RSIH-92		714'-11"	I		
RSIH-93		714'-11"	I		
RSIH-95		711'-2"	I		
RSIH-96		711'-2"	I		
RSIH-98		701'-2"	I		
RSIH-163		717'-9"	I		
RSIH-167		717'-9"	I		
RSIH-413 A&B		722'-8"	A		
RISH-414		716'-10"	I		
RISH-442		717'-9 1/2"	I		
RSIH-469		707'-6 1/2"	I		
RSIH-476		707'-1-3/4"	I		
SIRH-9		737'-0"	I		
SIRH-11		718'-6"	I		
SIRH-17		730'-0"	I		
SIRH-18		730'-0"	I		
SIRH-22		711'-4"	I		
SIRH-23 A&B		711'-4"	I		
SIRH-26		705'-0"	I		

Unit 1 - Amendment No. 1A, 49, 50, 63

Unit 2 - Amendment No. 8, 43, 44, 57

SAFETY RELATED SNUBBERS

<u>Snubber No.</u>	<u>Location</u>	<u>Elevation</u>	<u>Accessible or Inaccessible (A or I)</u>	<u>Snubbers Especially Difficult to Remove</u>	<u>In High Radiation Area During Shutdown</u>
<u>UNIT II</u>					
RCVCH-1396	Chemical & Vol	702'-10"	I		
RCVCH-1505	Control	708'-6"	I		
RCVCH-1513		710'-1"	I		
RCVCH-1524		719'-1"	I		
RCVCH-1574		721'-0"	I		
RCVCH-1668		705'-5"	I		
RCVCH-1373		722'-11"	I		
RCVCH-1389		706'-1"	I		
RRCH-253		704'-4"	I		
RRCH-255		704'-8"	I		
RRCH-251		707'-2"	I		
RRCH-288		707'-2"	I		
RRCH-291		704'-6"	I		
RRCH-292		704'-7"	I		
<u>UNIT I</u>					
CCH-304	Comp Cooling	717'-7"	A		
CCH-373		712'-4"	A		
CCH-376 A&B		700'-5"	A		
CCH-377		703'-0"	A		
CCH-378		708'-4"	A		
CCH-380		670'-8"	A		
CCH-381 A&B		671'-4"	A		
CCH-397		699'-3"	A		
CCH-398 A&B		671'-4"	A		
<u>UNIT II</u>					
CCH-161	Comp Cooling	717'-7"	A		
CCH-166		719'-11"	A		
CCH-167		720'-0"	A		
CCH-172		720'-0"	A		
CCH-173		708'-5"	A		
CCH-176		705'-3"	A		
CCH-179 A&B		671'-4"	A		
CCH-180		670'-8"	A		
CCH-181		708'-4"	A		
CCH-182		704'-2"	A		
CCH-185 A&B		671'-4"	A		
CCH-186		670'-10"	A		
<u>UNIT I</u>					
RCSH-81	Containment Spray	760'-9"	I		
RCSH-82		760'-8"	I		
RCSH-83 A&B		732'-1"	I		
<u>UNIT II</u>					
CSH-75 A&B	Containment Spray	731'-10"	I		
CSH-76		752'-7"	I		
CSH-79		751'-9"	I		
CSH-82 A&B		731'-11"	I		
CSH-83		767'-2"	I		
CSH-84		767'-2"	I		
CSH-210		698'-0"	I		
CSH-215		698'-0"	A		
CSH-224		710'-6"	A		

Unit 1 - Amendment No. 1A, A9, B0, 63
 Unit 2 - Amendment No. B, AB, AA, 57

SAFETY RELATED SNUBBERS

<u>Snubber No.</u>	<u>Location</u>	<u>Elevation</u>	<u>Accessible or Inaccessible (A or I)</u>	<u>Snubbers Especially Difficult to Remove</u>	<u>In High Radiation Area During Shutdown</u>
RRHH-20 RRHH-62	<u>UNIT I</u> RHR	704'-3"	A		
		705'-10"	A		
CVCRH-6 RRHH-21	<u>UNIT II</u> RHR	711'-0"	I		
		704'-6"	A		
ZX-PSCH-127	<u>UNIT II</u> ZX	707'-0"	A		

Unit 1 - Amendment No. 1A, 49, 63
 Unit 2 - Amendment No. 8, 43, 57

3.5 EVENT MONITORING INSTRUMENTATION

Applicability

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

Objectives

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

A. Specification - Process Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-1 shall be Operable.
2. With the number of Operable event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-1, either restore the inoperable channels to Operable status within seven days, or be in at least Hog Shutdown within the next 12 hours.
3. With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirements of Table TS.3.15-1, either restore the minimum number of channels to Operable status within 48 hours or be in at least Hot Shutdown within the next 12 hours.

B. Specification - Radiation Monitors

1. The event monitoring instrumentation channels specified in Table TS.3.15-2 shall be Operable.
2. With the number of Operable event monitoring instrumentation channels less than the Required Total Number of Channels shown on Table TS.3.15-2, either restore the inoperable channels to Operable status within seven days, or prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.7.B.2 within the next 30 days outlining the action taken, the cause of the inoperability, the plans and the schedule for restoring the system to Operable status.
3. With the number of Operable event monitoring instrumentation channels less than the Minimum Channels Operable requirement of Table TS.3.15-2, initiate the preplanned alternate method of monitoring the appropriate parameters in addition to submitting the report required in (2) above.

Unit 1 - Amendment No. 46, 63

Unit 2 - Amendment No. 40, 57

Basis

The operability of the event 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 Lessons Learned Task Force Status Report and Short Term Recommendations."

Unit 1 - Amendment No. 40, 63

Unit 2 - Amendment No. 40, 57

TABLE TS.3.15-1
EVENT MONITORING INSTRUMENTATION - PROCESS

<u>Instrument</u>	<u>Required Total No. of Channels</u>	<u>Minimum Channels Operable</u>
1. Pressurizer Water Level	2	1
2. Auxiliary Feedwater Flow to Steam Generators (One Channel Flow and One Channel Wide Range Level for Each Steam Generator)	2/steam gen	1/steam gen
3. Reactor Coolant System Subcooling Margin***	2	1
4. Pressurizer Power Operated Relief Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve, and One Channel Acoustic Sensor per Valve*)	2/valve	1/valve
5. Pressurizer Power Operated Relief Block Valve Position (One Common Channel Temperature, One Channel Limit Switch per Valve, and One Channel Acoustic Sensor per Valve*)	2/valve	1/valve
6. Pressurizer Safety Valve Position (One Channel Temperature per Valve and Common Acoustic Sensor**)	2/valve	1/valve

* - A common acoustic sensor provides backup position indication for each pressurizer power operated relief valve and its associated block valve.

** - The acoustic sensor channel is common to both valves. When operable, the acoustic sensor may be considered as an operable channel for each valve.

*** - Fully qualified input instrumentation is being installed in accordance with the NRC's TMI Action Plan. Until installation is completed, this function will be satisfied using the plant process computer.

DPR-42 Amendment No. 43, 63
DPR-60 Amendment No. 49, 57

TABLE TS.3.15-1

TABLE TS.3.15-2
EVENT MONITORING INSTRUMENTATION - RADIATION

<u>Instrument</u>	<u>Required Total No. of Channels</u>	<u>Minimum Channels Operable</u>
1. Containment Radiation Monitors (HI Range)	2	1
2. Steam Relief Activity Monitors	1/steam line	1/steam line
3. High Range Shield Building Ventilation Monitors	1	1

Unit 1 - Amendment No. 63
 Unit 2 - Amendment No. 57

TABLE TS.3.15-2

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

Unit 1 - Amendment No. 58, 59, 61, 63 Unit 2 - Amendment No. 49, 52, 55, 57	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
	27. Turbine Overspeed 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	S	NA	R	NA	FSAR page 3.2-56
	34. Reactor Gap Exhaust Air Temperature	S	NA	R	NA	FSAR page 5.4-2
	35a. 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
	b. Post-Accident Monitoring Radiation Instruments	D	K	M	NA	Includes all those in Table TS.3.15-2
	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 4 KV Safeguard Busses	NA	R	M	NA	
	39. Loss of Voltage 4 KV Safeguard Busses	NA	R	M	NA	

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. At least once each 18 months, subject each diesel engine to a thorough inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service.

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.