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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated April 26, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (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-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 76, 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

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 7, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iii	iii
129	129
130	130
132	132
133	133
134	134
164a	--
164b	--
164c	--
164d	--
190	190
232b	232b
232c	232c
236	236
237	237
241ff	241ff
241hh	241hh
241ii	241ii

Amendment No. ~~26~~, 76

LIMITING CONDITION FOR OPERATION

3.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the interrelated parameters of pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the peak suppression chamber pressure does not exceed design values in the event of a loss-of-coolant accident.

Specification:

a. The downcomers in the suppression chamber shall have a minimum submergence of three feet and a maximum submergence of four and one half feet whenever the reactor coolant system temperature is above 215F.

b. During normal power operation, the combination of primary containment pressure and suppression chamber bulk pool temperature shall be within the shaded area of

SURVEILLANCE REQUIREMENT

4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

Applicability:

Applies to the periodic testing of the pressure suppression system pressure and suppression chamber water temperature and level.

Objective:

To assure that the pressure suppression system pressure and suppression chamber water temperature and level are within required limits.

Specification:

a. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

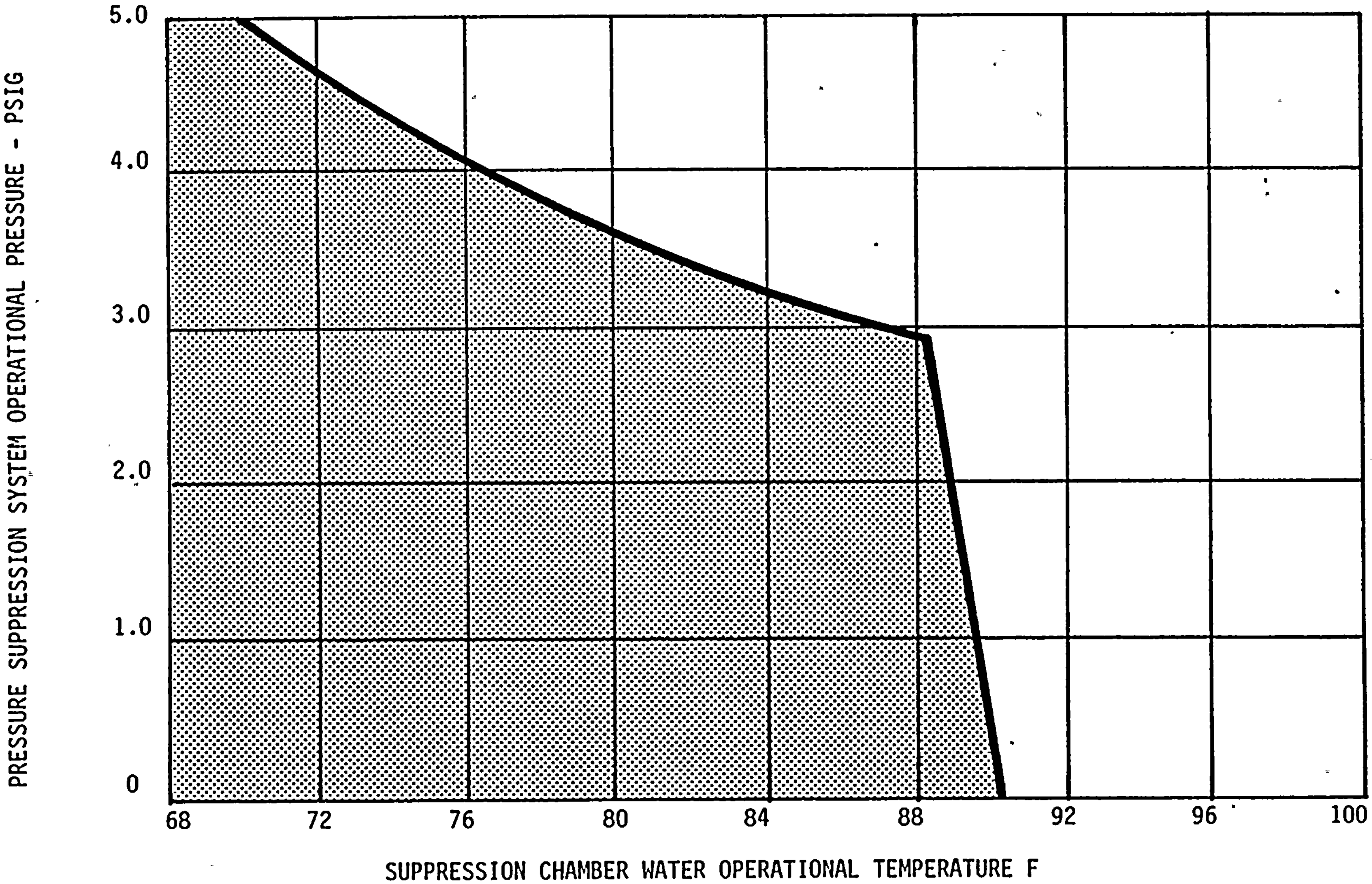
b. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(1) Figure 3.3.2a when downcomer submergence is greater than or equal to 4 feet, or (2) Figure 3.3.2b when downcomer submergence is greater than or equal to 3 feet but less than 4 feet. If these temperatures are exceeded, pool cooling shall be initiated immediately.	c. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.
c. If Specifications a and b above are not met within 24 hours, the reactor shall be shut down using normal shutdown procedures.	d. Whenever operation of a relief valve is indicated and the bulk suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 200 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.
d. During testing of relief valves which add heat to the torus pool, bulk pool temperature shall not exceed 10F above normal power operation limit specified in b above. In connection with such testing, the pool temperature must be reduced within 24 hours to below the normal power operation limit specified in b above.	e. Whenever there is indication of relief valve operation with the local temperature of the suppression pool reaching 200F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.
e. The reactor shall be scrammed from any operating condition when the suppression pool bulk temperature reaches 110F. Operation shall not be resumed until the pool temperature is reduced to below the normal power operation limit specified in b above.	
f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool bulk temperature reaches 120F.	

FIGURE 3.3.2 a

ALLOWABLE PRESSURE SUPPRESSION SYSTEM

4 FOOT DOWNCOMER SUBMERGENCE

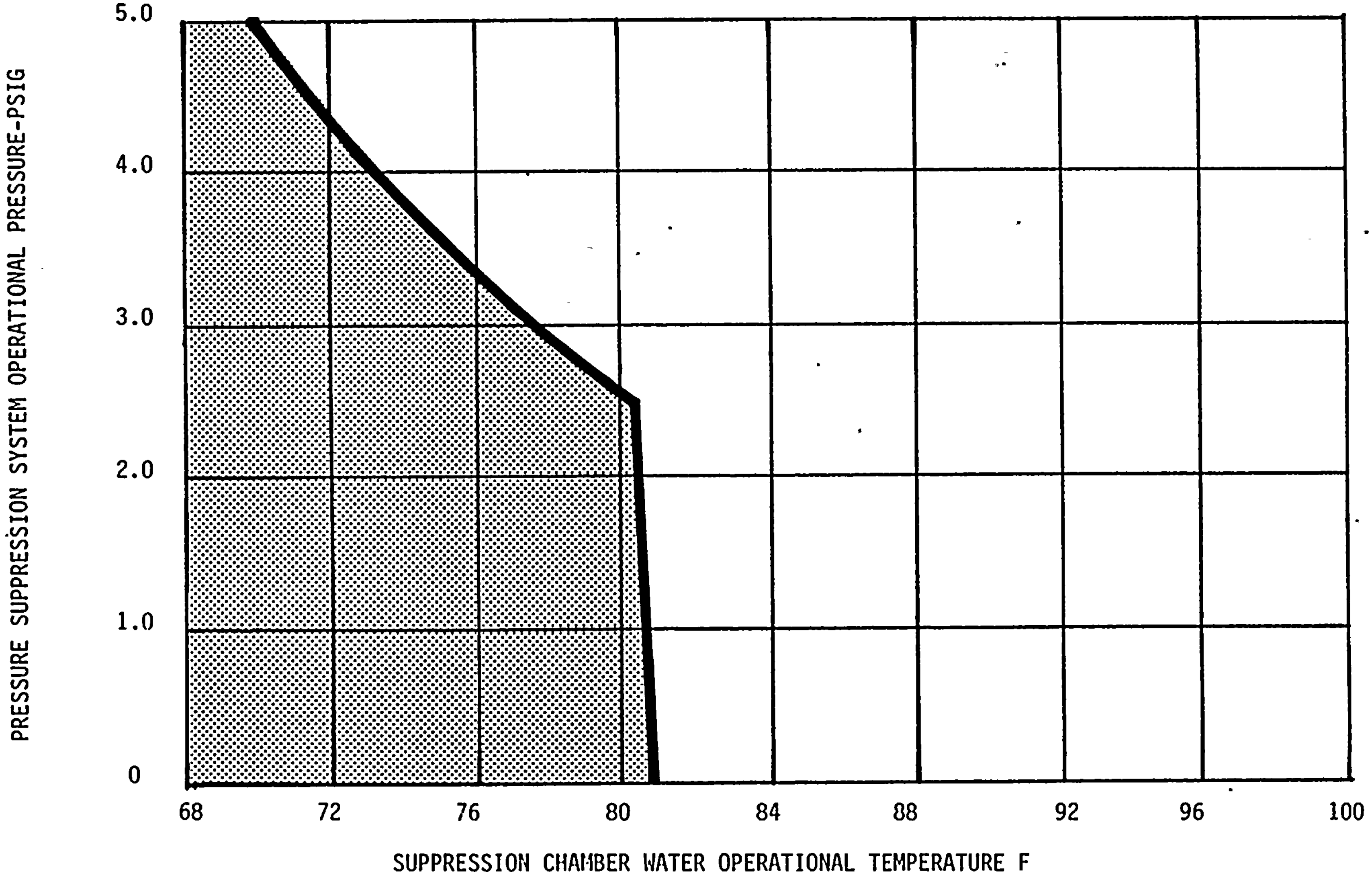


Amendment No. ~~26~~, 76

FIGURE 3.3.2 b

ALLOWABLE PRESSURE SUPPRESSION SYSTEM

3 FOOT DOWNCOMER SUBMERGENCE



Amendment No. ~~26~~, 76

BASES FOR 3.3.2 AND 4.3.2 PRESSURE SUPPRESSION SYSTEM PRESSURE AND SUPPRESSION CHAMBER WATER TEMPERATURE AND LEVEL

The values specified for suppression chamber water temperature, maximum downcomer submergence, and system pressures are based on the effect these parameters have on the short-term post-accident system pressure following a loss-of-coolant accident. The combinations shown on Figures 3.3.2 a and b and the water level required are based on maintaining the post-accident pressure below the design value of 35 psig and the maximum suppression chamber water temperature below 140F in the containment design basis loss-of-coolant accident (Appendix E-11.2.2.3).*

The calculational basis for the pressure suppression system initial conditions, Figures 3.3.2 a and b are presented in the Fifth Supplement.*

The three foot minimum and the four and one-half foot maximum submergence are a result of the Mark I Containment Long Term Program.

The 215F limit for the reactor is specified, since below this temperature the containment can tolerate a blowdown without exceeding the 35 psig design pressure of the suppression chamber without condensation.

Actually, for reactor temperatures up to 312F the containment can tolerate a blowdown without exceeding the 35 psig design pressure of the suppression chamber, without condensation.

Some experimental data suggests that excessive steam condensing loads might be encountered if the bulk temperature of the suppression pool exceeds 160F during any period of relief valve operation with sonic conditions at the discharge exit. This can result in local pool temperatures in the vicinity of the quencher of 200F. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the acti[illegible] to be taken in the event of a relief valve inadvertently opens or sticks open. As a minimum, this action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings

*FSAR

(8) Off-Gas and Vacuum Pump Isolation - The respective system shall be isolated orthe instrument channel shall be considered inoperable and Specification 3.6.1 shall be applied.

(9) Diesel Generator Initiation - The diesel generator shall be considered inoperable and Specification 3.6.3 shall be applied.

(10) Emergency Ventilation Initiation - The emergency ventilation system shall be considered inoperable and Specification 3.4.4 shall be applied.

(11) High Pressure Coolant Injection Initiation - The high pressure coolant injection system shall be considered inoperable and Specification 3.1.8.c shall be applied.

(12) Primary Containment Monitoring - The primary containment monitoring instrumentation shall be considered inoperable and Specification 3.3.2. shall be applied.

(13) Control Room Ventilation - The control room ventilation system shall be considered inoperable and Specification 3.4.5 shall be applied.

b. During operation with a Maximum Total Peaking Factor (MTPF) greater than the design value, either:

TABLE 3.6.21

PRIMARY CONTAINMENT MONITORING

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System	Setpoint	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	R[illegible]
(1) Suppression Chamber Water Level	2	1	Specification 3.3.2			X	X

TABLE 4.6.21

PRIMARY CONTAINMENT MONITORING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1) Suppression Chamber Water Level	once/day	N/A	Once Every Six Months

BASES FOR 3.6.2 AND 4.6.2 PROTECTIVE INSTRUMENTATION

The set points on the generator load rejection and turbine stop valve closure scram trips are set to anticipate and minimize the consequences of turbine trip with failure of the turbine bypass system as described in the bases for Specification 2.1.2. Since the severity of the transients is dependent on the reactor operating power level, bypassing of the scrams below the specified power level is permissible.

The primary containment monitoring system is provided to alert the operator of conditions which could reduce safety margins during a postulated Loss of Coolant Accident. Appropriate operator corrective action is described in Specification 3.3.2, should Limiting Conditions for Operation be exceeded. This monitoring instrumentation does not automatically initiate engineered safeguards systems.

Although the operator will set the setpoints at the values indicated in Tables 3.6.2.a-1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations include inherent instrument error, operator setting error and drift of the set point. These errors are compensated for in the transient analyses by conservatism in the controlling parameter assumptions as discussed in the bases for Specification 2.1.2. The deviations associated with the set points for the safety systems used to mitigate accidents have negligible effect on the initiation of these systems. These safety systems have initiation times which are orders of magnitude greater than the difference in time between reaching the nominal set point and the worst set point due to error. The maximum allowable set point deviations are listed below:

Neutron Flux
- APRM, +2.7% of rated neutron flux
- IRM, +2.5% of rated neutron flux

Recirculation Flow, ±1% of rated recirculation flow

Reactor Pressure, ±15.8 psig

Containment Pressure, ±0.053 psig

Reactor Water Level, ±2.6 inches of water

Main Steam Line Isolation Valve Position, ±2.5% of stem position

Scram Discharge Volume, +0 and -1 gallon

Condenser Low Vacuum, ±0.5 inches of mercury

High Flow-Main Steam Line, ± 1 psid

High Flow-Emergency Cooling Line, ± 1 psid

High Area Temperature-Main Steam Line, ± 10F

High Area Temperature-Clean-up and Shutdown, ± 6F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Emergency Cooling System Vent, +100% and -50% of set point

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

High Radiation-Offgas Line, ± 50% of set point, (Appendix D)*

Suppression Chamber Water Level, ± 1.8 inches

The test intervals for the trip systems result to calculated failure probabilities $\leq 10^{-4}$ which corresponds to the proposed IEEE Criteria for System Failure Probability. (IEEE SG-3, Information Docket #1 - Protection System Reliability, April 24, 1968).

The test intervals for the trip systems result in calculated failure probabilities ranging from 6.7×10^{-7} to 1.76×10^{-10} (Fifth Supplement, p. 115).* The more frequent sensor checks result in even less probability that the particular system will fail. Because of local high radiation, testing instrumentation in the area of the main steam line isolation valves can only be done during periods of Station shutdown. These functions include high area temperature isolation, high radiation isolation and isolation valve position scram.

Testing of the scram associated with the shutdown position of the mode switch can be done only during periods of Station shutdown since it always involves a scram.

*FSAR

TABLE 3.6.11-1

ACCIDENT MONITORING INSTRUMENTATION

Parameters	Total Number of Channels	Minimum Number of Operable Sensors or Channels	Action (See Table 3.6.11-2)
1) Relief Valve Position Indication	2/Valve	1/Valve	1
2) Safety Valve Position Indication	2/Valve	1/Valve	1
3) Reactor Vessel Water Level	2	1	2
4) Drywell Pressure Monitor	2	1	4
5) Suppression Chamber Water Level	2	1	4
6) Containment Hydrogen Monitor	2	1	4
7) Containment High Range Radiation Monitor	2	1	3
8) Suppression Chamber Water Temperature	2	1	2

TABLE 4.6.11

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENT

Parameter	Instrument Channel Test	Instrument Channel Calibration
(1) Relief valve position indicator (Primary - Acoustic)	Once per month	Once during each major refueling outage
Relief valve position indicator (Backup - Thermocouple)	Once per month	Once during each major refueling outage
(2) Safety valve position indicator (Primary - Acoustic)	Once per month	Once during each major refueling outage
Safety valve position indicator (Backup - Thermocouple)	Once per month	Once during each major refueling outage
(3) Reactor vessel water level	Once per month	Once during each major refueling outage
(4) Drywell Pressure Monitor	Once per month	Once during each major refueling outage
(5) Suppression Chamber Water Level Monitor	Once per month	Once during each major refueling outage
(6) Containment Hydrogen Monitor	Once per month	Once per quarter
(7) Containment High Range Radiation Monitor	Once per month	Once during each major refueling outage
(8) Suppression Chamber Water Temperature	Once per month	Once during each major refueling outage

BASES 3.6.11 and 4.6.11 ACCIDENT MONITORING INSTRUMENTATION

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 Lessons Learned Task Force Status Report and Short-Term Recommendations" and/or NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980 and NUREG-0661, "Safety Evaluation Report Mark I Containment Long Term Program."