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. 26
License No. DPR-63

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

   A. The applications for amendment by Niagara Mohawk Power Corporation (the licensee) dated May 19 and May 25, 1977, comply 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 applications, 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 Appendices A and B, as revised through Amendment No. 26, 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

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 29, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

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

Pages

iii
129
130
131
132
133
188
190
236
237

Add pages:

164a
164b
164c
164d
232b
232c

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 water 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.

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.

LIMITING CONDITION FOR OPERATION

(1) Figure 3.3.2a when downcomer submergence is $\geq$ 4 feet, or (2) Figure 3.3.2b when downcomer submergence is $\geq$ 3 feet. If these temperatures are exceeded, pool cooling shall be initiated immediately.

c. If Specifications a and b above are not met within 24 hours, the reactor shall be shutdown using normal shutdown procedures.

d. During testing of relief valves which add heat to the torus pool, the water temperature shall not exceed 10F above the normal power operation limit specific 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. The reactor shall be scrammed from any operating condition when the suppression pool 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 temperature reaches 120F.

SURVEILLANCE REQUIREMENT

d. Whenever operation of a relief valve is indicated and the 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.

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Figure 3.3.2 a
ALLOWABLE PRESSURE SUPPRESSION SYSTEM
4 FOOT DOWNCOMER SUBMERGENCE

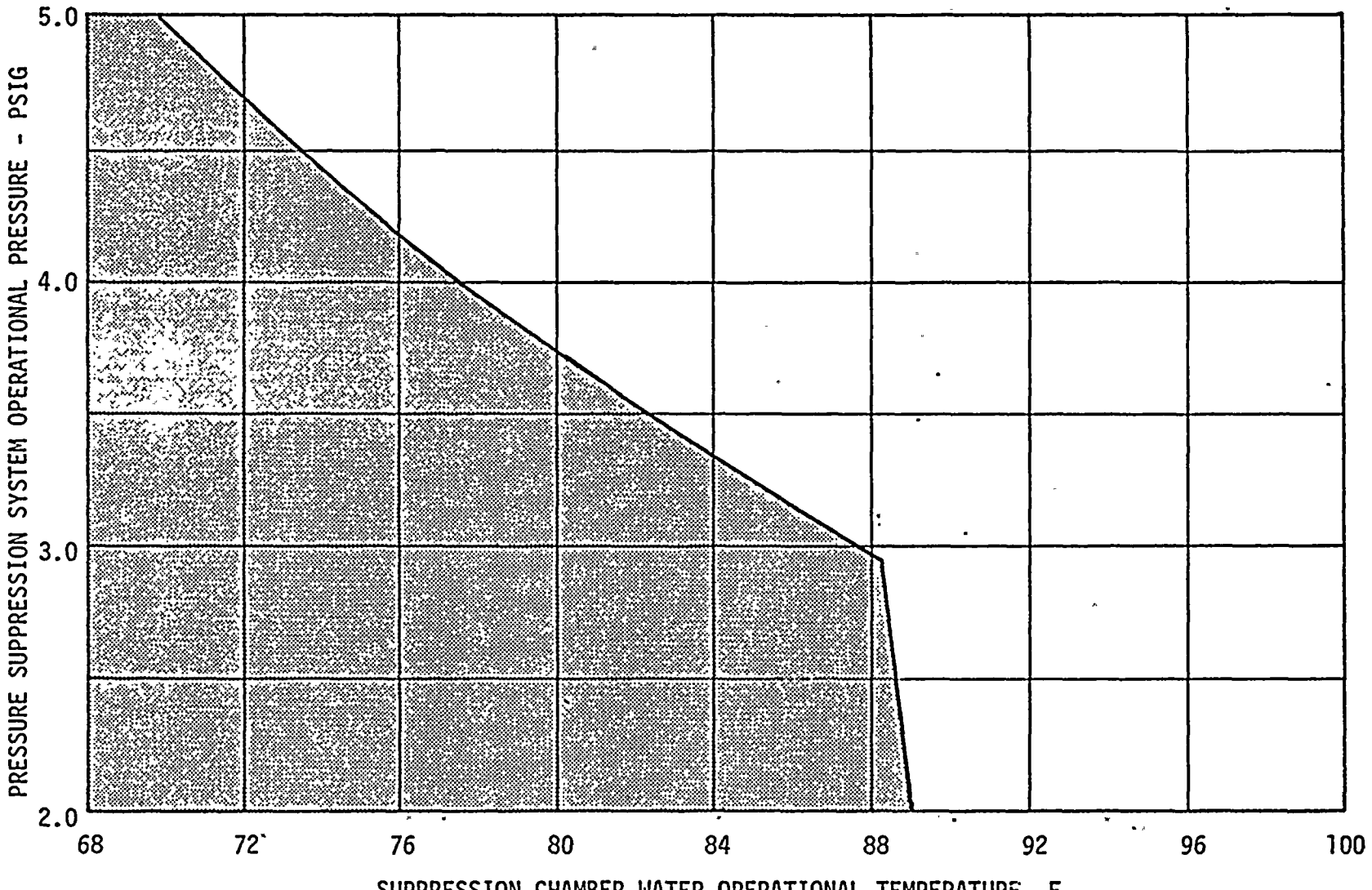
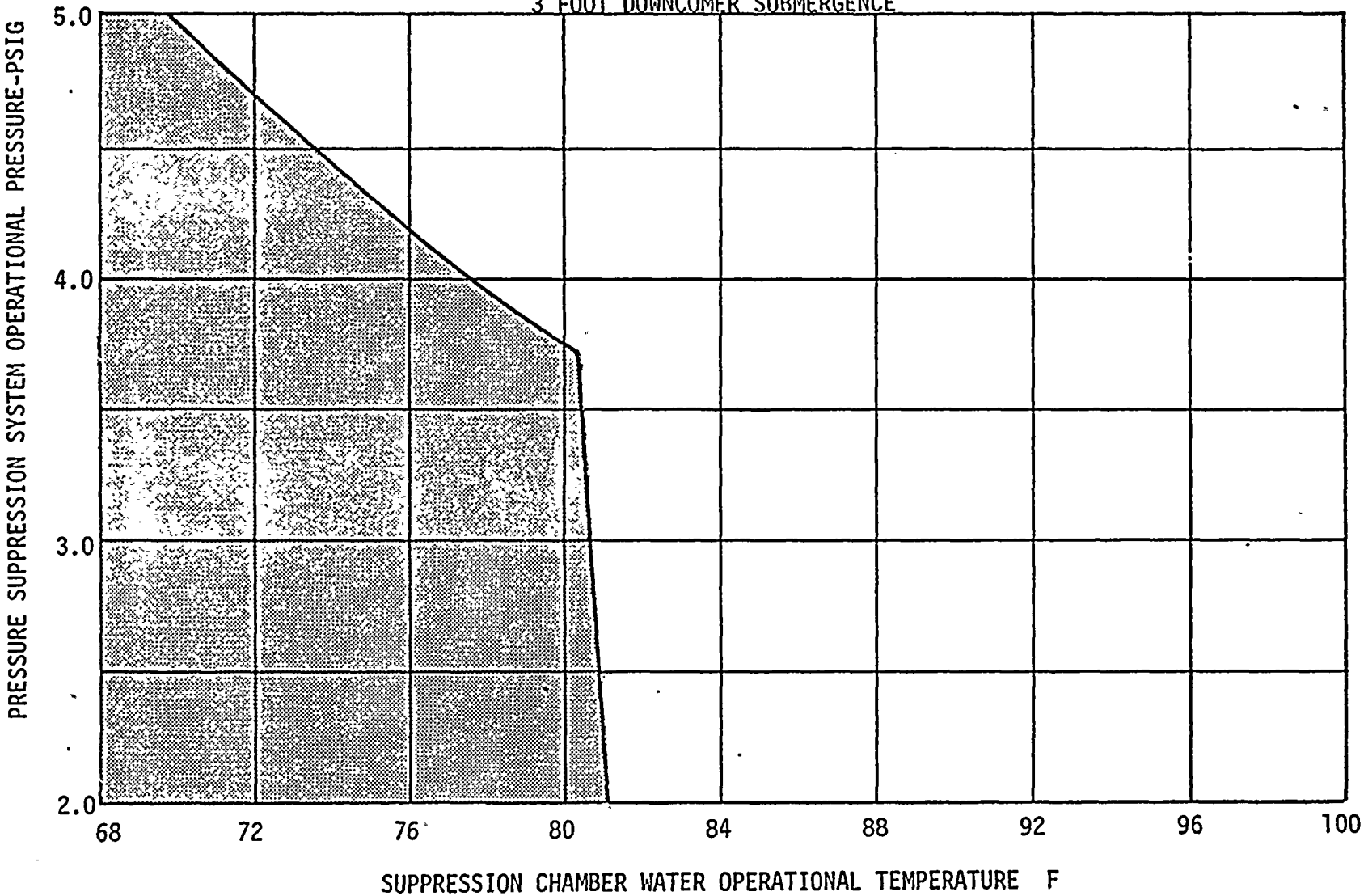


Figure 3.3.2 b
ALLOWABLE PRESSURE SUPPRESSION SYSTEM
3 FOOT DOWNCOMER SUBMERGENCE



LIMITING CONDITIONS FOR OPERATION

3.3.8 Drywell-Suppression Chamber Differential Pressure

Applicability:

Applies to the operational status of drywell-suppression chamber differential pressure system.

Objective:

To assure that the pressure suppression system will remain functional during a design basis loss-of-coolant accident.

Specification:

a. Differential pressure between the drywell and suppression chamber shall be maintained at or above levels according to Figure 3.3.8 except as specified in (1) and (2) below:

(1) The differential pressure shall be established within 24 hours of achieving operating pressure and temperature. The differential pressure may be reduced to less than that specified by Figure 3.3.8 24 hours prior to a scheduled shutdown.

(2) This differential may be decreased to less than that of Figure 3.3.8 for a maximum of four (4) hours during required operability testing of the drywell-pressure suppression chamber vacuum breakers.

SURVEILLANCE REQUIREMENTS

4.3.8 Drywell-Suppression Chamber Differential Pressure

Applicability:

Applies to the periodic testing requirements for the drywell-suppression chamber differential pressure system.

Objective:

To verify the operability of the drywell-suppression chamber differential system.

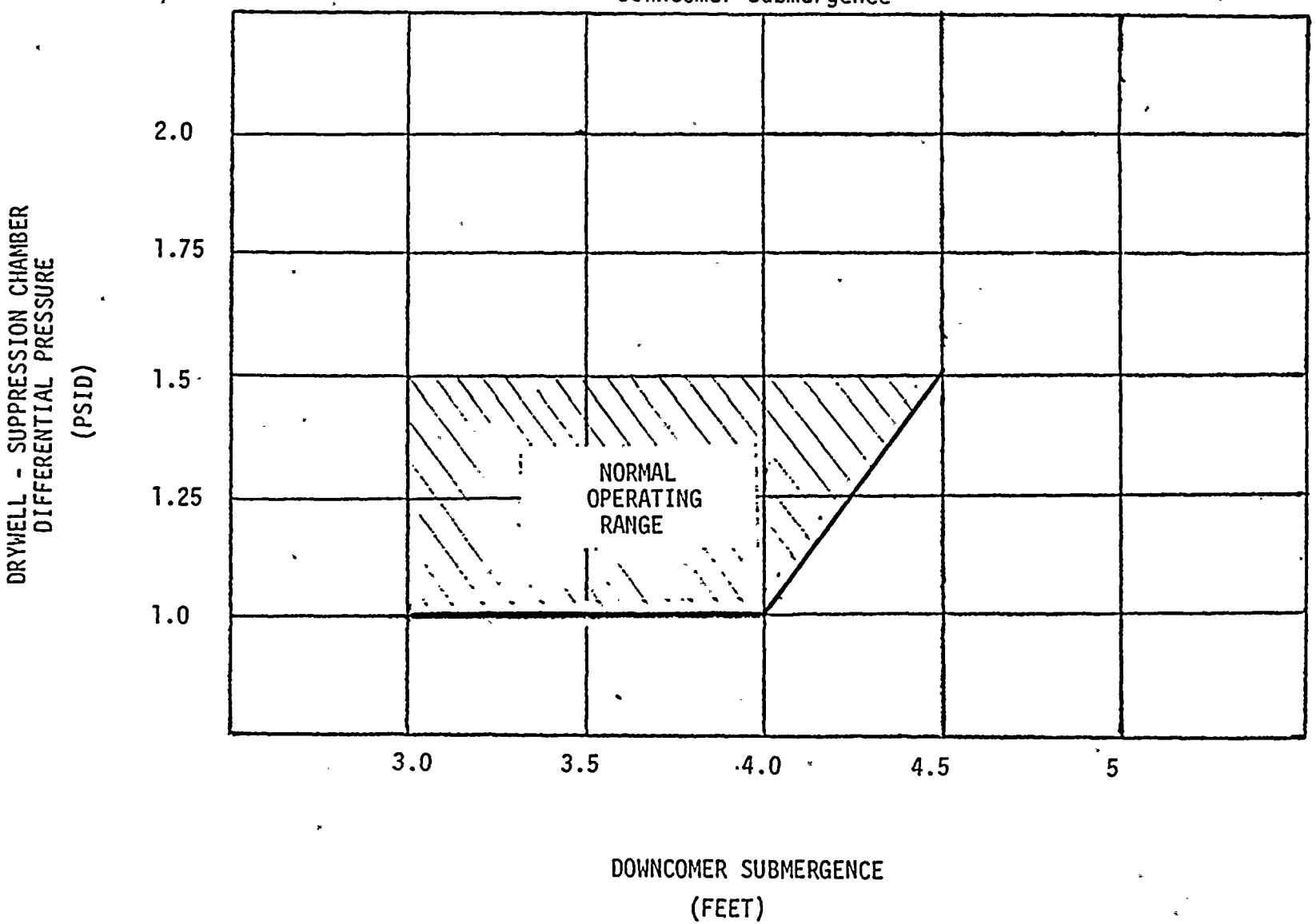
Specification:

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
b. If the differential pressure of Specification 3.3.8.a cannot be maintained and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition within the following 18 hours.	

FIGURE 3.3.8

Drywell-Suppression Chamber Differential Pressure
vs.
Downcomer Submergence



BASES FOR 3.3.8 AND 4.3.8 DRYWELL-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure in accordance with Figure 3.3.8 and suppression chamber water level corresponding to a downcomer submergence range of 3.0 to 4.5 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs a safety function.

Objective:

To assure the operability of the instrumentation required for safe operation.

Specification:

a. The set points, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.6.2a to 3.6.21.

If the requirements of a table are not met, the actions listed below for the respective type of instrumentation shall be taken.

(1) Instrumentation that initiates scram-control rods shall be inserted.

4.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs a safety function.

Objective:

To verify the operability of protective instrumentation.

Specification:

a. Sensors and instrument channels shall be checked, tested and calibrated at least as frequently as listed in Tables 4.6.2a to 4.6.21.

(8) Off-Gas and Vacuum Pump Isolation - The respective system shall be isolated or the 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 3.3.8 shall be applied.

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

(1) The APRM scram and rod block settings shall be reduced to the values given by the equations in Specification 2.1.2.a; or

(2) The power distribution shall be changed such that the MTPF no longer exceeds the design value.

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	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(1) Drywell-Suppression Chamber Differential Pressure	2	1	Figure 3.3.8			X	X
(2) 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)	Drywell-Suppression Chamber Differential Pressure	once/day	N/A	Once Every Six Months
(2)	Suppression Chamber Water Level	once/day	N/A	Once Every Six Months

L9

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 3.3.8, 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, $\pm$ 1% of rated recirculation flow

Reactor Pressure, $\pm$15.8 psig

Containment Pressure, $\pm$0.053 psig

Reactor Water Level, $\pm$2.6 inches of water

Main Steam Line Isolation Valve Position, $\pm$2.5% of stem position

Scram Discharge Volume, + 0 and - 1 gallon

Condenser Low Vacuum, $\pm$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)*

Drywell-Suppression Chamber Differential Pressure, ±0.1 psid

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