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

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. DPR-50

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensees), dated May 18, 1981, as supplemented and revised June 10, 1981, August 4, 1981, November 13, 1981, and May 20, 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.

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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-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. Unless the Commission modifies conditions of the Licensing Board's Partial Initial Decision dated December 14, 1981, which would affect this action, this license amendment becomes effective 60 days after the Commission accepts the Board's recommendations as set forth in that Decision, or restart, whichever is earlier.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 20, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 78

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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.

* Remove	Insert
ii	ii
iv	iv
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
Fig. 2.3-1	Fig. 2.3-1
3-6	3-6
--	3-6a
3-7	3-7
3-18c	3-18c
--	3-18d
3-25	3-25
3-25a	--
3-26	3-26
--	3-26a
3-28	3-28
3-29	3-29
3-30	3-30
3-31	3-31
3-32	3-32
3-32a	3-32a
3-37	3-37
3-37a	3-37a
--	3-40a
--	3-40b
--	3-40c
3-105	3-105
4-2	4-2
--	4-2a
4-5	4-5
4-5a	4-5a
4-7	4-7
--	4-7a
4-8	4-8
4-39	4-39
4-47	4-47
4-51	4-51**
4-52	4-52
--	4-52a
--	4-52b

*Changes on the revised TS pages do not become effective immediately (see Item 3 of the amendment). In the interim, the existing TS pages are in effect and should be retained.

**This page has not changed. It is included to maintain document completeness.

TABLE OF CONTENTS

Amendment No. ~~59~~, ~~72~~, 78

TABLE OF CONTENTS

Amendment No. ~~11~~, ~~28~~, ~~30~~, ~~41~~, ~~47~~, ~~65~~, ~~72~~, 78

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

Bases

The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure (2). The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established(2). The maximum settings for the reactor high pressure trip (2300 psig) and the pressurizer code safety valves (2500 psig)(3) have been established in accordance with ASME Boiler and Pressure Vessel Code, Section III, Article 9, Winter, 1968 to assure that the reactor coolant system pressure safety limit is not exceeded. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig(4).

References

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4

(4) FSAR, Table 4-1

c. Reactor coolant system pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip set point is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1 for high reactor coolant system pressure has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient (6). Due to calibration and instrument errors, the safety analysis assumed a 45 psi pressure error in the high reactor coolant system pressure trip setting.

The high pressure trip setpoint was subsequently lowered from 2390 psig to 2300 psig. The lowering of the high pressure trip setpoint and raising of the setpoint for the Power Operated Relief Valve (PORV), from 2255 psig to 2450 psig, has the effect of reducing the challenge rate to the PORV while maintaining ASME Code Safety Valve capability.

The low pressure (1800 psig) and variable low pressure (11.75 T_{OUT}-5103) trip setpoint were initially established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction (3,4). The B&W generic ECCS analysis, however, assumed a low pressure trip of 1900 psig and, to establish conformity with this analysis, the low pressure trip setpoint has been raised to the more conservative 1900 psig. Figure 2.3-1 shows the high pressure, low pressure, and variable low pressure trips.

d. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperature in the operating range.

The calibrated range of the temperature channels of the RPS is 520° to 620°F. The trip setpoint of the channel is 619°F. Under the worst case environment, power supply perturbations, and drift, the accuracy of the trip string is 1°F. This accuracy was arrived at by summing the worst case accuracies of each module. This is a conservative method of error analysis since the normal procedure is to use the root mean square method.

Therefore, it is assured that a trip will occur at a value no higher than 620°F even under worst case conditions. The safety analysis used a high temperature trip set point of 620°F.

The calibrated range of the channel is that portion of the span of indication which has been qualified with regard to drift, linearity, repeatability, etc. This does not imply that the equipment is restricted to operation within the calibrated range. Additional testing has demonstrated that in fact, the temperature channel is fully operational approximately 10% above the calibrated range.

Since it has been established that the channel will trip at a value of RC outlet temperature no higher than 620°F even in the worst case, and since the channel is fully operational approximately 10 % above the calibrated range and exhibits no hysteresis or foldover characteristics, it is concluded that the instrument design is acceptable.

e. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

f. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to value $\leq$ 5.0 percent of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$ 5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

References

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

(6) Technical Specification Change Request No. 31, January 16, 1976, and Technical Specification Change Request No. 84, June 23, 1978.

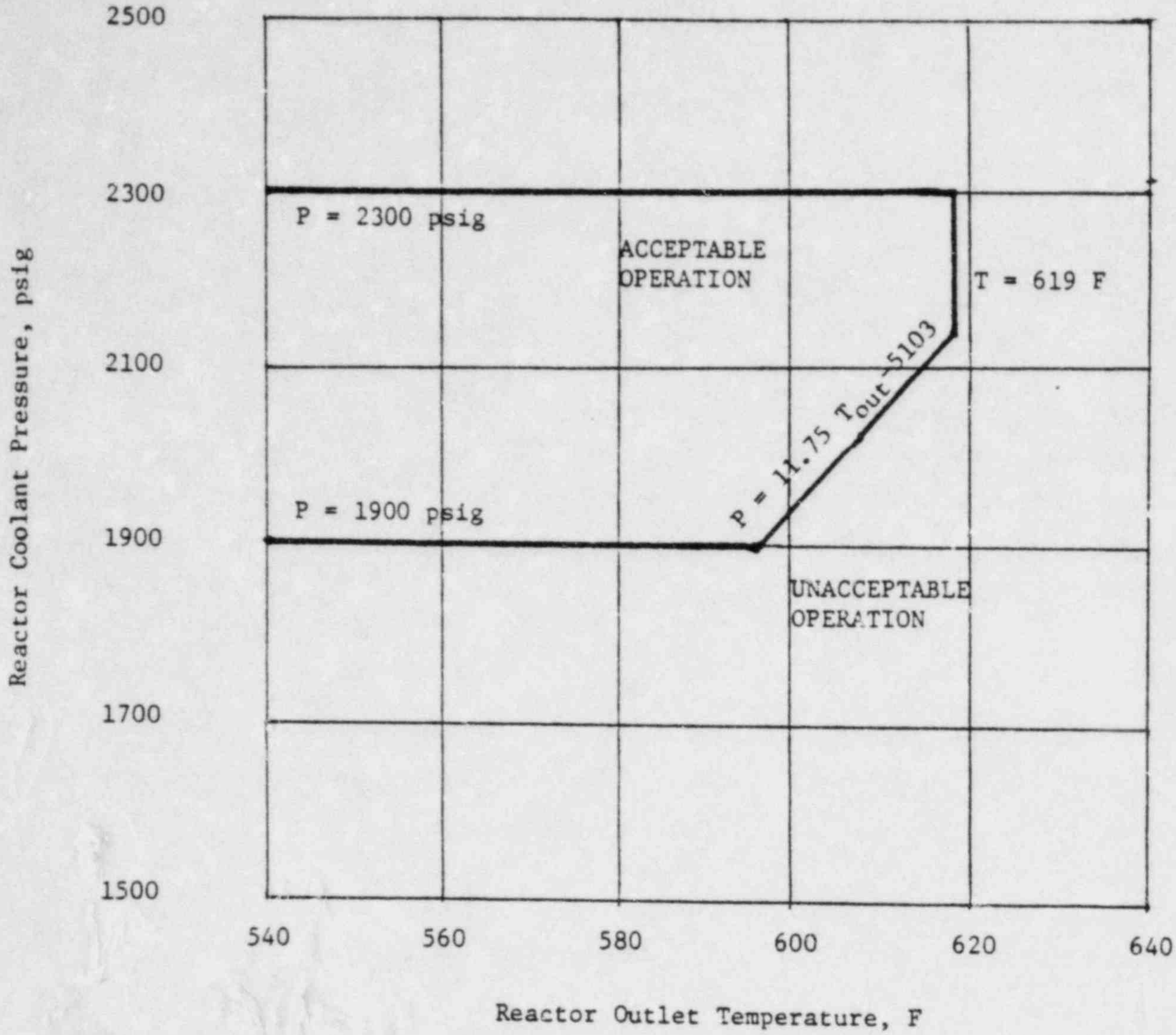
(7) "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," BAW-10103, Rev. 2, Babcock and Wilcox, April 1976.

TABLE 2.3-1

REACTOR PROTECTION SYSTEM TRIP SETTING LIMITS

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, Max. % of rated power	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow (2) and imbalance max. of rated power	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3. Nuclear power based (5) on pump monitors, Max. % of rated power	NA	NA	91%	Bypassed
4. High reactor coolant system pressure, psig max.	2300	2300	2300	1720(4)
5. Low reactor coolant system pressure, psig min.	1900	1900	1900	Bypassed
6. Variable low reactor coolant system pressure psig, min.	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	(11.75 Tout-5103)(1)	Bypassed
7. Reactor coolant temp. F., Max.	619	619	619	619
8. High Reactor Building pressure, psig, max.	4	4	4	4

(1) Tout is in degrees Fahrenheit (F)
(2) Reactor coolant system flow, %
(3) Administratively controlled reduction set only during reactor shutdown
(4) Automatically set when other segments of the RPS (as specified) are bypassed
(5) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation
(6) Trip settings limits are setting limits on the setpoint side of the protection system bistable connectors.



TMI-1
PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 2.3-1

3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

Applicability

Applies to reactor coolant system conditions required prior to criticality.

Objective

a. To limit the magnitude of any power excursions resulting from reactivity insertion due to moderator pressure and moderator temperature coefficients.

b. To assure that the reactor coolant system will not go solid in the event of a rod withdrawal or startup accident.

c. To assure sufficient pressurizer heater capacity to maintain natural circulation conditions during a loss of offsite power.

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requiements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT +10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 Pressurizer

3.1.3.4.1 The reactor shall be maintained subcritical by at least one percent $\Delta k/k$ until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer.

(a) **With the pressurizer level outside the required band, be in at least HOT SHUTDOWN with the reactor trip breakers open within 6 hours and be in COLD SHUTDOWN within an additional 30 hours.**

3.1.3.4.2 A minimum of 107 kw of pressurizer heaters, from each of two pressurizer heater groups shall be OPERABLE. Each OPERABLE 107 kw of pressurizer heaters shall be capable of receiving power from a 480 volt ES bus via the establisned manaul transfer scheme.

(a) With the pressurizer inoperable due to one (1) inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

(b) With the pressurizer inoperable due to two (2) inoperable emergency power supplies to the pressurizer heaters either restore the inoperable emergency power supplies within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

3.1.3.5 Safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality with the following exceptions:

a. Inoperable rod per 3.5.2.2.

b. Physics Testing per 3.1.9.

c. Shutdown margin may not be reduced below 1% Δ k/k per 3.5.2.1.

d. Exercising rods per 4.1.2.

Following safety rod withdrawal, the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to deboration.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525°F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525 F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT+ 10°F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The availability of at least 107 kw in pressurizer heater capability is sufficient to maintain primary system pressure assuming normal system heat losses. Emergency power to heater groups 8 or 9, supplied via a manual transfer scheme, assures redundant capability upon loss of offsite power.

The requirements that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirements for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3.

(2) FSAR, Section 3.2.2.1.4

3.1.12 Pressurizer Power Operated Relief Valve (PORV) and Block Valve

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.1 The PORV shall not be taken out of service, nor shall it be isolated from the system (except that the PORV may be isolated to limit leakage to within the limits of specification 3.1.6) unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out or MU-V16A/B/C/D and MU-V217 are closed.
b. Head of the Reactor Vessel is removed.
c. T_{avg} is above 320°F.

3.1.12.2 The PORV settings shall be as follows, within the tolerances of ± 25 psi and ± 12°F:

Above 275°F - 2450 psig
Below 275°F - 485 psig

3.1.12.3 If the reactor vessel head is installed and T_{avg} is ≤275°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D and MU-V217 are closed, and
b. Pressurizer level is ≤220 inches.

3.1.12.4 PORV and Block Valve

The PORV and the associated block valve shall be OPERABLE during HOT STANDBY, STARTUP, AND POWER OPERATION:

a. With the PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the PORV block valve inoperable, within 1 hour either restore the PORV block valve to OPERABLE status or close the PORV (verify closed) and remove power from the PORV; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With either the PORV or block valve inoperable, restore the operable valve to operable status prior to startup from the next cold shutdown unless the cold shutdown occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a cold shutdown occurs within this 90-day period, restore the inoperable valve to operable status prior to the startup for the next fuel cycle.

Bases

If the PORV is removed from service, sufficient measures are incorporated to prevent overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere. In order to prevent exceeding leakage rates specified in T.S. 3.1.6., the PORV may be isolated.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2.

With RCS temperatures less than 275°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than 220 inches to prevent overpressurization in the event of any single failure.

Both the PORV and the PORV block valve should be operable during the HOT STANDBY, STARTUP, AND POWER OPERATION. If either the PORV or the PORV block valve are inoperable, the PORV discharge line should be isolated to prevent potential uncontrolled RCS depressurization.

3.4 DECAY HEAT REMOVAL - TURBINE CYCLE

Applicability

Applies to the operating status of equipment that functions to remove decay heat, utilizing the secondary side of the Steam Generators.

Objective

To define the conditions necessary to assure immediate availability of the Emergency Feedwater (EFW) System and Main Steam Safety Valves.

Specification

3.4.1 With the Reactor Coolant System temperature greater than 250°F, three independent EFW pumps and associated flow paths* shall be OPERABLE with:

a. Two EFW pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW pump capable of being powered from an OPERABLE steam supply system.

b. With one pump or flow path* inoperable, restore the inoperable pump or flow path to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 12 hours. With more than one EFW pump or flow path* inoperable, restore the inoperable pumps or flow paths* to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

c. Four of six turbine bypass valves are OPERABLE.

d. The condensate storage tanks (CST) shall be OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. With a CST inoperable, restore the CST to operability within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 30 hours. With more than one CST inoperable, restore the inoperable CST to OPERABLE status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

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* For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator

3.4.2 With the reactor coolant system temperature greater than 250°F, all eighteen (18) main steam safety valves shall be OPERABLE or, if any are not OPERABLE, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when RCS temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If Main Steam Safety Valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.3, such that the remaining safety valves can accommodate the decay heat.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust. (1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.1, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1 assure that before the reactor is heated to above 250 ^{o} F, adequate auxiliary feedwater capacity is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm,each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident.

REFERENCES

(1). FSAR Section 10.2.1.3.

reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and four high reactor building pressure instrument channels. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 20% reactor power. The safety features actuation system must have two analog channels functioning correctly prior to startup. The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column "B" (Table 3.5-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR Section 7.

There are four reactor protection channels. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other instrumentation channels is one out of two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is by-passed. There will be one reactor protection system bypass switch key permitted in the control room.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 460 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Eight hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

REFERENCE

FSAR, Section 7.1

TABLE 3.5-1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. Reactor Protection System			
1. Manual pushbutton	1	0	Maintain hot shutdown
2. Power range instrument channel	3(a)	1(a)	Maintain hot shutdown
3. Intermediate range instrument channels	1	0	Maintain hot shutdown (b)
4. Source range instrument channels	1	0	Maintain hot shutdown (c)
5. Reactor coolant temperature instrument channels	2	1	Maintain hot shutdown
6. Pressure-temperature instrument channels	2	1	Maintain hot shutdown
7. Flux/imbalance/flow instrument channels	2	1	Maintain hot shutdown
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	Maintain hot shutdown
b. Low reactor coolant pressure instrument channels	2	1	Maintain hot shutdown

Table 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. Reactor Protection System (con't.)			
9. Power/number of pumps instrument channels	2	1	Maintain hot shutdown
10. High reactor building pressure channels	2	1	Maintain hot shutdown

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, or 2 of 4 power range instrument channels are greater than 10 percent full power, hot shutdown is not required.

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
B. Other Reactor Trips			
1. Loss of Feedwater	2(a)	1(a)	Maintain less than 7% indicated reactor power.
2. Turbine Trip	2(b)	1(b)	Maintain less than 20% indicated reactor power.

(a) Bypass of the feedwater pump trip signal may be placed in effect when indicated reactor power is less than 7%. The bypass will be removed when reactor power is raised above 7%.

(b) The main turbine trip bypass may be placed in effect when indicated reactor power is less than 20%. The bypass will be removed when the reactor power is increased above 20%.

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A)	(B)	(C)
C. Engineered Safety Features	Minimum Operable Channels	Minimum Degree of Redundancy	Operator Action if Conditions of Column A and B Cannot be met(a)
1. Makeup and Purification System (high pressure injection mode)			
a. Reactor Coolant Pressure Instrument Channels	2	1	Hot Shutdown
b. Reactor Building 4 psig Instrument Channels	2	1	Hot Shutdown
c. Manual Pushbutton (b)	2	1	Hot Shutdown
2. Decay Heat System (low pressure injection mode)			
a. Reactor Coolant Pressure Instrument Channels	2	1	Hot Shutdown
b. Reactor Building 4 psig Instrument Channels	2	1	Hot Shutdown
c. Reactor Coolant Pressure D.H. Valve Interlock Bistable	1	0	Open circuit breaker at MCC for DH-V1 or DH-V2 with the affected valve in the closed position or maintain R.C. pressure less than 350 psig.

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met (a)
C. Engineered Safety Features (con't.)			
3. Reactor Building Isolation and Reactor Building Cooling System			
a. Reactor Bldg. 4 psig (g) Instrument Channel	2	1	Hot Shutdown
b. Manual Pushbutton	2	1	Hot Shutdown
c. RPS Trip	2	1	Hot Shutdown
d. Reactor Building 30 psig	2	1	Hot Shutdown
e. RCS Pressure less than 1600 psig	2	1	Hot Shutdown
f. Reactor Bldg. Purge line Isolation (AHV-1A and AHV-1D) High Radiation	1	0	(f)
4. Reactor Building Spray System			
a. Reactor Building 30 psig Instrument Channel	2(d)	1	Hot Shutdown
b. Spray Pump Manual Switches (c)	2	1	Hot Shutdown
5. 4.16KV ES Bus Undervoltage Relays			
a. Degraded Grid Voltage Relays	2	1	(e)
b. Loss of Voltage Relay	2	1	(e)
6. Emergency Feedwater System (all pumps auto start)			
a. Loss of both Feedwater Pumps	2	1	Hot Shutdown
b. Loss of all RC Pumps	2	1	Hot Shutdown

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Unit

C. Engineered Safety Features (con't.)

(a) If minimum conditions are not met within 24 hours, the unit shall then be placed in a cold shutdown condition.

(b) Also initiates Low Pressure Injection

(c) Spray valves opened by manual pushbutton listed in Item 3 above.

(d) Two out of three switches in each actuation channel operable.

(e) If a relay fails in the untripped state, it shall be placed in a tripped state within 12 hours to obtain a degree of redundancy of 1. The relay may be removed from the tripped state for up to 2 hours for functional testing pursuant to Table 4.1-1.

(f) Discontinue Reactor Building purging and close AHV-1A, 1B, 1C, and 1D. Note: (a) above does not apply if AHV-1A, 1B, 1C and 1D are closed.

(g) For hot functional testing, prior to Cycle 5 criticality the 4 psig signal is not required for Nuclear Service Closed Cycle Cooling water, Intermediate Cooling and Reactor Coolant Pump Seal Injection (return line only). Two operable channels of a 30 psig Reactor Building isolation signal with a minimum degree of redundancy of 1 are required if the 4 psig signal is not operable for these lines.

3.5.3 ENGINEERED SAFEGUARDS PROTECTION SYSTEM ACTUATION SETPOINTS

Applicability:

This specification applies to the engineered safeguards protection system actuation setpoints.

Objective:

To provide for automatic initiation of the engineered safeguards protection system in the event of a breach of Reactor Coolant System integrity.

Specification:

3.5.3.1 The engineered safeguards protection system actuation setpoints and permissible bypasses shall be as follows:

Initiating Signal	Function	Setpoint
High Reactor Building Pressure (1)	Reactor Building Spray	$\leq$ 30 psig
	Reactor Building Isolation	$\leq$ 30 psig
	High-Pressure Injection	$\leq$ 4 psig
	Low-Pressure Injection	$\leq$ 4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	$\leq$ 4 psig*
Low Reactor Coolant System Pressure	High Pressure Injection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Low Pressure Injection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Reactor Building Isolation	$\geq$ 1600 psig (2)
4.16 kv E.S. Buses Undervoltage Relays		
Degraded Voltage (5)	Switch to Onsite Power Source and load shedding	3595 volts (4)
Degraded grid timer		10 sec (5)
Loss of voltage	Switch to Onsite Power Source and load shedding	2400 Volts (6)
Loss of voltage timer		1.5 sec (7)

(1) May be bypassed for reactor building leak rate test.
(2) May be bypassed below 1725 psig on decreasing pressure and is automatically reinstated before 1800 psig on increasing pressure.
(3) May be bypassed below 875 psig on decreasing pressure and is automatically reinstated before exceeding 950 psig on increasing pressure.

(4) Minimum allowed setting is 3560 v. Maximum allowed setting is 3650 v.

(5) Minimum allowed time is 8 sec. maximum allowed time is 12 sec.

(6) Minimum allowed setting is 2200 volts, maximum allowed setting is 2860 volts.

(7) Minimum allowed time is (1.0) second, maximum allowed time is (2.0) seconds.

*For Hot Functional Testing prior to Cycle 5 criticality, the 4 psig Reactor Building isolation signal is not required for Nuclear Service Closed Cycle Cooling water, Intermediate cooling water and Reactor Coolant Pump seal injection (return line only). Remote Manual and 30 psig Reactor Building isolation signals are required if the 4 psig signal is not operable for these lines.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a LOCA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressures to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1600 and 500 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum to break sizes and is far enough below normal operating pressure to prevent spurious initiation. Bypass of HPI below 1725 psig and LPI below 825 psig, prevents ECCS actuation during normal system cooldown.

4.16 KV ES Bus Undervoltage Relays

The basis for the degraded grid voltage relay setpoint is to protect the safety related electrical equipment from loss of function in the event of a sustained degraded voltage condition on the offsite power system. The timer setting prevents spurious transfer to the onsite source for transient conditions.

The loss of voltage relay and timers detect loss of offsite power condition and initiate transfer to the onsite source with minimal time delay.

3.5.5 ACCIDENT MONITORING INSTRUMENTATION

Applicability

Applies to the operability requirements for the instrument identified in Table 3.5-2 during STARTUP, POWER OPERATION and HOT STANDBY.

Objective

To assure operability of key instrumentation useful in diagnosing situations which could lead to inadequate core cooling.

Specification

3.5.5.1 The minimum number of channels identified for the instruments in Table 3.5-2, shall be OPERABLE. With the number of instrumentation channels less than the minimum required, restore the inoperable channel(s) to OPERABLE status within seven (7) days (48 hours for pressurizer level) or be in at least HOT SHUTDOWN within the next six (6) hours and in COLD SHUTDOWN within an additional 30 hours. Prior to startup following a COLD SHUTDOWN, the minimum number of channels shown in Table 3.5-2 shall be OPERABLE.

Bases

The saturation Margin Monitor provides a quick and reliable means for determination of saturation temperature margins. Hand calculation of saturation pressure and saturation temperature margins can be easily and quickly performed as an alternate indication for the Saturation Margin Monitors.

Discharge flow from the two (2) pressurizer code safety valves and the PORV is measured by differential pressure transmitters connected across elbow taps downstream of each valve. A delta-pressure indication from each pressure transmitter is available in the control room to indicate code safety or relief valve line low. An alarm is also provided in the control room to indicate that discharge from a pressurizer code safety or relief valve is occurring. In addition, an acoustic monitor is provided to detect flow in the PORV discharge line. An alarm is provided in the control room for the acoustic monitor.

The Emergency Feedwater System is provided with two channels of flow instrumentation on each of the two discharge lines. Local flow indication is also available for the emergency feedwater system through the Non-Nuclear Instrumentation (NNI).

Although the pressurizer has multiple level indications, the separate indications are selectable via a switch for display on a single display. Pressurizer level, however, can also be determined via the patch panel and the computer log. In addition, a second channel of pressurizer level indication is available independent of the NNI.

Although the instruments identified in Table 3.5-2 are significant in diagnosing situations which could lead to inadequate core cooling, loss of any one of the instruments in Table 3.5-2 would not prevent continued, safe, reactor operation. Therefore, operation is justified for up to 7 days (48 hours for pressurizer level). Alternate indications are available for Saturation Margin Monitors using hand calculations, the PORV/Safety Valve position monitors using discharge line thermocouple and Reactor Coolant Drain Tank indications, and for EFW flow using Steam Generator level and EFW pump discharge pressure. Pressurizer level has two channels, one channel from NNI (3 D/P instrument strings through a single indicator) and one channel independent of the NNI. Operation with the above channels out of service is permitted for up to 48 hours. Alternate indication would be available through the plant computer.

TABLE 3.5-2

ACCIDENT MONITORING INSTRUMENTS

FUNCTION	INSTRUMENTS	NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS
1	Saturation Margin Monitor	2	1
2	Safety Valve Differential Pressure Monitor	1 per discharge line	1 per discharge line
3	PORV Position Monitors	2	1*
4	Emergency Feedwater Flow	2 per flow path	1 per flow path
5	Pressurizer Level	2	1

* With the PORV Block Valve closed in accordance with Specification 3.1.12.4.a, the minimum number of channels is zero.

TABLE NOTATION

*At all times.
**During waste gas holdup system operation.
***Operability is not required when discharges are positively controlled through the closure of WDG-V47, and RM-A8 and FT-151 are operable.

ACTION 25 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank may be released to the environment for up to 14 days provided that prior to initiating the release:

1. At least two independent samples of the tank's contents are analyzed, and

2. At least two technically qualified members of the Unit staff independently verify the release rate calculations and verify the discharge valve lineup.

3. The Manager Unit 1 shall approve each release.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 26 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 28 days provided the flow rate is estimated at least once per 4 hours.

ACTION 27 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 28 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours. (See also Specification 3.5.1 Table 3.5-1, Item 3.f.)

ACTION 30 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement:

†1. a. Within 4 hours, take and analyze an initial grab sample. Grab samples shall be taken every 72 hours until the monitor is declared operable.

b. If the sample indicates greater than 2% oxygen or 3% hydrogen, within one hour begin to add nitrogen to reduce the concentration to less than 2% oxygen or 3% hydrogen.

c. Prior to Startup when deborating from a shutdown boron concentration in anticipation of going critical except during physics testing, if the hydrogen concentration is greater than 3%, nitrogen shall be added to reduce the hydrogen concentration to 3%. During Startup operations as described above, a grab sample shall be taken and analyzed every 4 hours.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protection channels is required once every four weeks on a rotational or perfectly staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protection channels is as follows:

Channels A, B, C, & D	Before Startup, when shutdown greater than 24 hours
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protection channels shall be trip tested prior to startup when the reactor has been shutdown for greater than 24 hours. Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.

REFERENCE

(1) FSAR, Section 7.1.2.3.4

CHANNEL DESCRIPTIONS	CHECK	TEST	CALIBRATE	REMARKS
19. Reactor Building Emergency Cooling and Isolation System Channels				
a. Reactor Building 4 psig Channels	S(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
b. RCS Pressure 1600 psig	S(1)	M(1)	NA	(1) When RCS Pressure >1800 psig
c. RPS Trip	S(1)	M(1)	NA	(1) When CONTAINMENT INTEGRITY is required
d. Reactor Bldg. 30 psig	S(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
e. Reactor Bldg. Purge Line High Radiation (AH-V-1A/D)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
f. Line Break Isolation Signal (ICCW & NSCCW)	W(1)	M(1)	R	(1) When CONTAINMENT INTEGRITY is required
20. Reactor Building Spray System Logic Channel	NA	Q	NA	
21. Reactor Building Spray System Analog Channels				
a. Reactor Building 30 psig Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R	(1) Check with Relative Position Indicator
24. Control Rod Absolute Position	S(1)	NA	R	(1) Check with Absolute Position Indicator
25. Core Flooding Tanks				
a. Pressure Channels	S(1)	NA	R	(1) When Reactor Coolant system pressure is greater than 700 psig
b. Level Channels	S(1)	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Makup Tank Level Channels	D(1)	NA	R	(1) When Makeup and Purification System Is In operation

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
28.	Radiation Monitoring Systems	W(1)(3)	M(3)	Q(2)	(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that the monitor is functioning. (2) Except area gamma radiation monitors RM-G6, RM-G7, and RM-G8 which are located in high radiation areas of the Reactor Building. These monitors will be calibrated quarterly or at the next scheduled reactor shutdown following the quarter in which calibration would normally be due, if a shutdown during the quarter does not occur. (3) Surveillances are to be performed only when containment integrity is required. This applies to monitors which initiate containment isolation only.
29.	High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
38.	Steam Generator Water Level	W	NA	R	
39.	Turbine Overspeed Trip	NA	R*	NA	
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R	
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R	
42.	Diesel Generator Protective Relaying	NA	NA	R	
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)				
	a. Degraded Grid	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons
	b. Loss of Voltage	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Reactor Trip	S(1)	M(1)	R	(1) When reactor power exceeds 7% power
46.	Turbine Trip/Reactor Trip	S(1)	M(1)	R	(1) When reactor power exceeds 20% power
47.a	Pressurizer Code Safety Valve and PORV tailpipe flow monitors	S(1)		R	(1) When T_{ave} is greater than 525°F
.b	PORV - Acoustic/Flow	NA	M(1)	R	(1) When T_{ave} is greater than 525°F
48.	PORV Setpoints	NA	M(1)	R	(1) Per Specification 3.1.12 excluding valve operation.

*Test to be performed prior to exceeding 20% power during Cycle 5 startup only.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
49.	Saturation Margin Monitor	S(1)	M(1)	R	(1) When T_{ave} is greater than 525°F.
50.	Emergency Feedwater Flow Instrumentation	NA	M(1)	R	(1) When T_{ave} is greater than 250°F.
51.	Emergency Feedwater Initiation				
	a. Loss of RCP's	NA	Q(1)(2)	R	(1) When T_{ave} is greater than 250°F.
	b. Loss of both Feedwater Pumps	NA	Q(1)(2)	R	(2) Includes logic test only

S - Each Shift
D - Daily
W - Weekly
M - Monthly

T/W - Twice per week
B/M - Every 2 months
Q - Quarterly
P - Prior to each startup if not done previous week

R - Each Refueling Period
NA - Not applicable
B/W - Every two weeks

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rods	Rod drop times of all full length rods	Each refueling shutdown
2.	Control Rod Movement	Movement of each rod	Every two weeks, when reactor is critical
3.	Pressurizer Safety Valves	Setpoint*	50% each refueling period
4.	Main Steam Safety Valves	Setpoint	25% each refueling period***
5.	Refueling System Interlocks	Functional	Start of each refueling period
6.	Main Steam Isolation Valves	(See Section 4.8)	
7.	Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8.	Deleted		
9.	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10.	Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation- Visual inspection of Intake Pump House Floor	Each refueling period
		(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11.	Pressurizer Block Valve (RC-V2)	Functional**	Quarterly

* The setpoint of the pressurizer code safety valves shall be in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968.

** Function shall be demonstrated by operating the valve through one complete cycle of full travel.

*** The required percentage of Main Steam Safety Valves will be tested prior to Cycle 5 criticality.

4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Loading Sequence

Applicability:

Applies to periodic testing requirements for safety actuation systems.

Objective:

To verify that the emergency loading sequence and automatic power transfer is operable.

Specifications:

4.5.1.1 * Sequence and Power Transfer Test

a. During each refueling interval, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable.

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

-M. U. Pump
-D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
-R. B. Cooling Pump
-R. B. Ventilators
-D. H. Closed Cycle Cooling Pump
-N. S. Closed Cycle Cooling Pump
-D. H. River Cooling Pump
-N. S. River Cooling Pump
-D. H and N. S. Pump Area Cooling Fan
-Screen House Area Cooling Fan
-Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psig Pressure Test Signal.)
-Motor Driven Emergency Feedwater Pump

c. Following successful transfer to the emergency diesel, the diesel generator breaker will be opened to simulate trip of the generator then reclosed to verify block load on the reclosure.

* This test shall be performed prior to Cycle 5 criticality.

4.5.1.2 Sequence Test

a. At intervals not to exceed 3 months, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power.

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

Bases

The Emergency loading sequence and automatic power transfer controls the operation

d. The battery will be subjected to a load test at a frequency not to exceed refueling periods. The battery voltage as a function of time will be monitored to establish that the battery performs as expected during this load test.

4.6.3 Pressurizer Heaters

a. The following tests shall be conducted at least once each refueling:

(1) Pressurizer heater groups 8 and 9 shall be transferred from the normal power bus to the emergency power bus and energized. Upon completion of this test, the heaters shall be returned to their normal power bus.

(2) Demonstrate that the pressurizer heaters breaker on the emergency bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

(3) Verify that following input of the Engineered Safeguards Signal, the circuit breakers, supplying power to the manually transferred loads for pressurizer heater groups 8 and 9, have been tripped.

Bases

The tests specified are designed to demonstrate that one diesel-generator will provide power for operation of safeguards equipment. They also assure that the emergency generator control system and the control systems for the safeguards equipment will function automatically in the event of a loss of normal a-c station service power or upon the receipt of an engineered safeguards Actuation Signal. The automatic tripping of manually transferred loads, on an Engineered Safeguards Actuation Signal, protects the diesel generators from a potential over-load condition. The testing frequency specified is intended to identify and permit correction of any mechanical or electrical deficiency before it can result in a system failure. The fuel oil supply, starting circuits, and controls are continuously monitored and any faults are alarmed and indicated. An abnormal condition is these systems would be signaled without having to place the diesel generators on test.

Precipitous failure of the station battery is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails.

The PORV has a remotely operated block valve to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valve and the block valves is supplied from an ESF power source to ensure the ability to seal this possible RCS leakage path.

The requirement that a minimum of 107 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation.

REFERENCE

(1) FSAR, Section 8.2

4.8 MAIN STEAM ISOLATION VALVES

Applicability

Applies to the periodic testing of the main steam isolation valves.

Objective

To specify the minimum frequency and type of tests to be applied to the main steam isolation valves.

Specification

4.8.1 A check of valve stem movement, up to 10 percent, shall be performed on a monthly basis when the unit is operational and under normal flow and load conditions.

4.8.2 The main steam isolation valves shall be tested at intervals not to exceed the normal refueling outage. Closure time of approximately 112 seconds shall be verified. This test will be performed under no flow and no load conditions.

Bases

Since a portion of the main steam lines and the steam lines to the main feed pump turbines are located in the turbine hall which is not protected against hypothetical tornado, missile, or aircraft incident; main steam isolation stop check valves are provided and located in the hardened portion of the intermediate building. These stop check valves are remotely closed by the operator from the control room, close in less than two minutes, and are tight closing [1] for long term containment isolation. Their ability to close upon signal should be verified at intervals not to exceed each scheduled refueling shutdown, and valve stem freedom should be checked on a monthly basis.

References

(1) FSAR, Section 10.2.1.3

4.9 EMERGENCY FEEDWATER SYSTEM PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine driven and two motor-driven Emergency feedwater pumps associated actuation signal, and valves.

Objective

To verify that the Emergency Feedwater (EFW) System is capable of performing its design function.

Specification

4.9.1 TEST

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the EFW pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3210. The test frequency shall be at least every 31 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 During testing of the EFW System when the reactor is in STARTUP, HOT STANDBY or POWER OPERATION, if one steam generator flow path* is made inoperable, a dedicated qualified individual who is in communication with the control room shall be continuously stationed at the EFW local manual valves (See Table 4.9-1). On instruction from the Control Room Operator, the individual shall realign the valves from the test mode to their operational alignment.

4.9.1.3 At least once per 31 days each valve listed in Table 4.9-1 shall be verified to be in the status specified in Table 4.9-1, when required to be operable.

4.9.1.4 On a quarterly basis, verify that the manual control (HIC-849/850) valve station functions properly.

4.9.1.5 On a quarterly basis, EFV-30A and 30B shall be checked for proper operation by cycling each valve over its full stroke.

4.9.1.6 Prior to start-up, following a refueling shutdown or a cold shutdown greater than 30 days, conduct a test to demonstrate that the motor driven EFW pumps can pump water from the condensate tanks to the Steam Generators.

*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a Steam Generator

4.9.2 ACCEPTANCE CRITERIA

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

Bases

The 31-day testing frequency will be sufficient to verify that the turbine driven and two motor-driven EFW pumps are operable and that the associated valves are in the correct alignment. ASME Section XI Article IWP-3210 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal acceptance criteria assures that the EFW pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. In the case of the EFW System flow, the flow shall be considered acceptable if under the worst case single pump failure, a minimum of 500 gpm can be delivered when steam generator pressure is 1050 psig and one steam generator is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient flow can be delivered to either Steam Generator. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Table 4.9-1

Status of EFW Valves

Valve No.	Status
CO-V-10A	Open
CO-V-10B	Open
EF-V-1A	Open
EF-V-1B	Open
EF-V-2A	Open
EF-V-2B	Open
MSV-2A	Open
MSV-2B	Open
EF-V4	Locked Closed
EF-V5	Locked Closed
EF-V6*	Locked Open
EF-V10A*	Locked Open
EF-V10B*	Locked Open
EF-V-16A*	Locked Open
EF-V-16B*	Locked Open
EF-V-20A*	Locked Open
EF-V-20B*	Locked Open
EF-V-22	Locked Open
CO-V-176	Locked Open

*Manual valve to which Specification 4.9.1.2 applies