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.

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3.1.3.5 Safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality with the following exceptions: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

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_{ave} is above 320°F.

3.1.12.2 The PORV settings shall be as follows, within the tolerances of $\pm$ 25 psi and $\pm$ 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_{ave} is $\leq$275°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16 A/B/C/D and MU-V217 are closed, and
b. Pressurizer level is $\leq$ 220 inches.

3.1.12.4 PORV and Block Valve

The PORV and the associated block valve shall be OPERABLE during HOT STANDBY, START UP, 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 lea[illegible] 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.

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.

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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.

3.4.3 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 STANDBY 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 dacay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to the condenser as fewedwater in the steam generator is coverted 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 accomodate 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 3.4.2, 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 to 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 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 permited 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.

<u>REFERENCE</u>

FSAR. Section 7.1

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Units	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A Cannot be Met
Reactor Protection System			
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
9. Power/number of pumps instrument channels	2	1	Maintain hot shutdown
10. High reactor building pressure channels	2	1	Maintain hot shutdown
Other Reactor Trips			
1. Loss of Feedwater	2[1]	1[1]	Maintain less than 7% indicated power
2. Turbine Trip	2[2]	1[2]	Maintain less than 20% indicated power

1. 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%.

2. The main turbine trip bypass may be placed in effect when indicated 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 Engineered Safety Features	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A cannot be met (a)
3. Reactor Building Isolation and Reactor Building Cooling System			
a. Reactor Building 4 psig 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

(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.

4. Reactor Building Spray System			
a. Reactor Building 30 psig Instrument Channel	2 (b)	1	Hot Shutdown
b. Spray Pump Manual Switches (c)	2	1	Hot Shutdown

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

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

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

TABLE 3.5-1 Continued

INSTRUMENTS OPERATING CONDITIONS

Functional Unit Engineered Safety Features	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A cannot be met (a)
5. Emergency Feedwater System			
Loss of Feedwater or RCP Pump (all four) - Start Motor and Turbine Pumps	2	1	Hot Shutdown.

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

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	4 psig
	Low-Pressure Injection	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 Presure Iniection	$\geq$ 1600(2) and $\geq$ 500(3) psig
	Reactor Building Isolation	$\geq$ 1600 psig (2)

(1) May be bypassed for reactor building leak rate test.

(2) May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

(3) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoint 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 pressure 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 of break sizes and is far enough below normal operating pressure to prevent spurious initiation. Bypass of HPI below 1750 psig, and LPI below 900 psig, prevents ECCS actuation during normal system cooldown.

3.5.5 ACCIDENT MONITORING INSTRUMENTATION

Applicability

Applies to the operability requirements for the instrument identified in Table 3.5-2 during START UP or POWER OPERATION.

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 or alternate indications, identified for the instruments in Table 3.5-2, shall be OPERABLE. With the number of instrumentation channels less than the minimum required and the alternate indication inoperable, restore the inoperable channel(s) to OPERABLE status within seven (7) days or be in at least HOT SHUTDOWN within the next twelve (12) hours. Prior to start-up 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. The hand calculation of saturation pressure and saturation temperature margins can be easily and quickly performed since it only requires knowledge of RC System loop temperatures and system pressure, and the use of steam tables; accordingly, hand calculation provides a suitable alternate indication for the Saturation Margin Monitor.

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 flow. An alarm is also provided in the control room to indicate that discharge from a pressurizer code safety or relief valve is occuring. 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.

In the event that a delta-pressure monitor or the acoustic monitor becomes inoperable, access to the containment would most likely be required; however, a reactor shutdown to allow containment access for this repair is not justifiable due to the existence of alternate means of detecting and monitoring code safety or relief valve discharge flow. The alternate means include discharge line thermocouples and Reactor Coolant Drain Tank indications.

The Emergency Feedwater System is provided with two channels of flow istrumentation on each of the two discharge lines. Local flow indication is also available for the emergency feedwater system.

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.

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 provided that the alternate indication is operable. Loss of an instrument and its alternate indication would degrade the reactor operators diagnostic capability and, thus, should be restored within seven (7) days.

TABLE 3.5-2

ACCIDENT MONITORING INSTRUMENTS

FUNCTION	INSTRUMENTS	NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS	ALTERNATE INDICATION
1	Saturation Margin Monitor	1	1	*
2	Safety/Relief Valve Differential Pressure Monitor	1 per discharge line	1 per discharge line	Discharge line thermocouples or Reactor Coolant Drain Tank level and pressure
3	PORV Acoustic Monitor	1	1	Discharge line thermocouples or Reactor Coolant Drain Tank level and pressure
4	Emergency Feedwater Flow	2 per flow path	1 per flow path	Steam Generator water level and EFW pump discharge pressure
5	Pressurizer Level	1	1	Pressurizer Level (patch panel) or Computer Log

* If the Saturation Margin Monitor is inoperable, the operability requirement for the Saturation Margin Monitor is satisfied by manual determination saturation temperature margins.

3.19 Separation of TMI-1 and TMI-2

Applicability

Applies to interconnections between TMI-1 and TMI-2 which have the potential for transferring significant quantities of contamination between units.

Objective

To control the transfer of radioactivity from TMI-2 to TMI-1 via system interties.

Specification

3.19.1 The isolation devices for the system interconnections listed in Table 3.19-1 shall remain inplace unless written approval has been received from the NRC. If approval for use of interconnections is received, use shall proceed under preestablished procedures.

3.19.2 No additional TMI-1/TMI-2 interconnections, with the potential of transferring significant quantities of radioactivity, shall be created without prior NRC approval.

3.19.3 Observed defeat of an isolation device, which separates a tieline between Units 1 and 2, without prior NRC approval shall be reported to the NRC as a special report within 30 days.

Bases

Interconnections exist between TMI-1 and TMI-2 that have the potential for transferring contamination to TMI-1 as a result of restoration activities at TMI-2. These interconnections should remain isolated unless approval for their use is received from the NRC.

Table 3.19-1

TMI-1/TMI-2 Interconnections

(1) Unit 2 Reactor Coolant Bleed Holdup Tank - Unit 1 Reactor Coolant Waste Evaporator

(2) Unit 1 Miscellaneous Waste Evaporator - Unit 2 Evaporator Condensate Test Tanks

(3) Unit 2 Neutralizer Tanks, Contaminated Drain Tanks, Reactor Coolant Bleed Holdup Tanks, Auxiliary Building Sump Tanks and Miscellaneous Waste Holdup Tanks - Unit 1 Liquid Waste Disposal System

(4) Unit 1 Evaporator Concentrate - Unit 2 Evaporator Concentrate

(5) Unit 1 Spent Ion Exchange Resin - Unit 2 Spent Ion Exchange Resin

Channel 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

TABLE 4.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

	CHANNEL DESCRIPTION	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 $>$ 1750 psig
	c. RPS Trip	S(1)	M(1)	NA	(1) When CONTAINMENT INTEGRITY is required
	d. Reactor Building 30 psig	S(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. Presure 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.	Makeup 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
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)	NA	M(1)	R	(1) Relay operation will be checked by local test push-buttons.
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 10% 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
47.b	PORV - Acoustic/Flow	NA	M	R	(1) When T_{ave} is greater than 525°F.
48.	PORV Setpoint	NA	M(1)	R	(1) When T_{ave} is greater than 525°F. Excluding valve operation.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
49.	Saturation Margin Monitor	S(1)	M(1)	R(1)	(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 or Feedwater	NA	Q(1)(2)	R	(1) When T_{ave} is greater than 250°F. (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

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	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 (b) Silt Accumulation Measurement of Pump House Floor	Each refueling period Quarterly
11.	Pressurizer Block Valve (RC-V2)	Functional**	Quarterly

* The set point 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.

Table 4.1-2 (Continued)

Item	Test	Frequency
12. Isolation devices on Unit 1/Unit 2 interconnections	Visual Inspection	Semiannually
(a) Unit 2 Reactor Coolant Bleed Holdup Tank - Unit 1 Reactor Coolant Waste Evaporator		
(b) Unit 1 Miscellaneous Waste Evaporator - Unit 2 Evaporator Condensate Test Tanks		
(c) Unit 2 Neutralizer Tanks, Containment Drain Tanks, Reactor Cooland Bleed Holdup Tanks, Auxiliary Building Sump Tanks and Miscellaneous Waste Holdup Tanks - Unit 1 Liquid Waste Disposal System		
(d) Unit 1 Evaporator Concentrate - Unit 2 Evaporator Concentrate		
(e) Unit 1 Spent Ion Exchange Resin - Unit 2 Spent Ion Exhange Resin		

4.5 EMERGENCY LDOADING 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 psi Pressure Test Signal.)
- Motor Driven Emergency Feedwater Pump

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.

c. Each time data are recorded, new data shall be compared with old to detect signs of abuse or deterioration.

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.

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 abiity 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

4.9 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 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 manual valves. 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:

(a). each EFW pump starting logic actuates upon receipt of an EFW test signal, and

(b). 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 cold shutdown for refueling, 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.

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. If testing indicates that the flow and/or pump head for a particular pump is not within the normal acceptance standard an evaluation of the pump performance shall be completed within 96 hours or the pump declared inoperable. For the case of the EFW System, the system shall be considered operable 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 generation 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

EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability:

Applies to periodic testing and surveillance requirement of the emergency power system.

Objective:

To verify that the emergency power system will respond promptly and perperly when required.

Specification:

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiated start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the nameplate rating (3000 kw). This test will be conducted every month on each diesel generator. Normal plant operation will not be affected.

b. Automatically start and loading the emergency diesel generator in accordance with specification 4.5.1.1 b/c including the following:

(1). Verify that the diesel generator starts from ambient condition upon receipt of the ES signal and is ready to load in $\leq$ 10 seconds.

(2.). Verify that the diesel block loads upon simulated loss of off-site power in $\leq$ 30 seconds.

(3). The diesel operates with the permanently connected and auto connected load for $\geq$ 5 minutes.

(4). The diesel engine does not trip when the generator breaker is opened while carrying emergency loads.

(5). The diesel generator block loads and operates for $\geq$ 5 minutes upon reclosure of the diesel generator breaker.

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

(7). That following input of the Engineered Safeguard Signal, it shall be verified that the circuit breakers, supplying power to the manually transferred loads for pressurizer heater Groups 8 and 9, have been tripped.

c. Each diesel generator shall be given an inspection at least annually in accordance with the manufacturer's recommendations for this class of standby service.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded monthly.

b. The voltage and specific gravity of a pilot cell will be measured and recorded monthly.