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 set point. The trip setting limit 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 RTD 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.

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 established manual transfer scheme.

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 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 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.

TABLE 3.5-1 (con't.)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit C. Engineered Safety Features con't.	(A) Minimum Operable Analog Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met (a)
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 redundance 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.

3.5.5 ACCIDENT MONITORING INSTRUMENTATION

Applicability

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

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 separation 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 situation 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.

INTENTIONALLY DELETED

INTENTIONALLY DELETED

TABLE 4.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

	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-A/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
49.	Saturation Margin Monitor	S(1)	M(1)	R	(1) When T_{ave} is greater than 525^{o}F.
50.	Emergency Feedwater Flow Instrumentation	NA	M(1)	R	(1) When T_{ave} is greater than 250^{o}F.
51.	Emergency Feedwater Initiation				
	a. Loss of RCP's	NA	A(1)(2)	R	(1) When T_{ave} is greater than 250^{o}F.
	b. Loss of both Feedwater Pumps	NA	Q(1)(2)	R	(2) Includes logic test only

S - Each Shift	T/W - Twice per week	R - Each Refueling Period
D - Daily	B/M - Every 2 months	NA - Not applicable
W - Weekly	Q - Quarterly	B/W - Every two weeks
M - Monthly	P - Prior to each startup if not done previous week	

4.6.3 Pressurizer Heaters

4.6.3.1 Once Each Refueling

a. 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.

b. Determine that the pressurizer heaters breaker on the emergency bus cannot be closed until the safeguards signal is bypassed and can be closed following bypass.

c. Determine that following input of the Engineered Safeguards 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.

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 dn 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. 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. 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