3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objectives

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

These conditions relate to: operational components, heatup and cooldown, leakage, reactor coolant activity, oxygen and chloride concentrations, minimum temperature for criticality, and reactor coolant system overpressure mitigation.

A. Operational Components

Specifications

1. Reactor Coolant Pumps

   a. A reactor shall not be brought critical with less than two pumps, in non-isolated loops, in operation.

b. If an unscheduled loss of one or more reactor coolant pumps occurs while operating below 10% rated power (P-7) and results in less than two pumps in service, the affected plant shall be shutdown and the reactor made subcritical by inserting all control banks into the core. The shutdown rods may remain withdrawn.

c. When the average reactor coolant loop temperature is greater than 350°F, the following conditions shall be met:

   1. At least two reactor coolant loops shall be operable.

   2. At least one reactor coolant loop shall be in operation.

d. When the average reactor coolant loop temperature is less than or equal to 350°F, the following conditions shall be met:

   1. A minimum of two non-isolated loops, consisting of any combination of reactor coolant loops or residual heat removal loops, shall be operable, except as specified in Specification 3.10.A.6.

   2. At least one reactor coolant loop or one residual heat removal loop shall be in operation, except as specified in Specification 3.10.A.6.

e. Reactor power shall not exceed 50% of rated power with only two pumps in operation unless the overtemperature ΔT trip setpoints have been changed in accordance with Section 2.3, after which power shall not exceed 60% with the inactive loop stop valves open and 65% with the inactive loop stop valves closed.

f. When all three pumps have been idle for > 15 minutes, the first pump shall not be started unless: (1) a bubble exists in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

2. Steam Generator

A minimum of two steam generators in non-isolated loops shall be operable when the average reactor coolant temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. One valve shall be operable whenever the head is on the reactor vessel, except during hydrostatic tests.

b. Three valves shall be operable when the reactor coolant average temperature is greater than 350°F, the reactor is critical, or the Reactor Coolant System is not connected to the Residual Heat Removal System.

c. Valve lift settings shall be maintained at 2485 psig ± 1 percent.

4. Reactor Coolant Loops

Loop stop valves shall not be closed in more than one loop unless the Reactor Coolant System is connected to the Residual Heat Removal System and the Residual Heat Removal System is operable.

5. Pressurizer

a. The reactor shall be maintained subcritical by at least 1% until the steam bubble is established and necessary sprays and at least 125 Kw of heaters are operable.

b. With the pressurizer inoperable due to inoperable pressurizer heaters either restore the inoperable heaters within 72 hours or be in at least hot shutdown within the next 6 hours.

c. With the pressurizer otherwise inoperable, be in at least hot shutdown with the reactor trip breakers open within 6 hours.

6. Relief Valves

a. Two power operated relief valves (PORVs) and their associated block valves shall be operable whenever the reactor keff is ≥0.99.

b. With one or more PORVs inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

c. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Basis

Specification 3.1.A-1 requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling flow in the event of a loss of reactor coolant flow accident. This provided flow will maintain the DNBR above 1.30.(1) Heat transfer analyses also show that reactor heat equivalent to approximately 10% of rated power can be removed with natural circulation; however, the plant is not designed for critical operation with natural circulation or one loop operation and will not be operated under these conditions.

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the reactor coolant system volume in approximately one half hour.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The requirement for redundant coolant loops ensures the capability to remove core decay heat when the reactor coolant system average temperature is less than or equal to 350°F. Because of the low-low steam generator water level reactor trip, normal reactor criticality cannot be achieved without water in the steam generators in reactor coolant loops with open loop stop valves. The requirement for two operable steam generators, combined with the requirements of Specification 3.6, ensure adequate heat removal capabilities for reactor coolant system temperatures of greater than 350°F.

Each of the pressurizer safety valves is designed to relieve 295,000 lbs. per hr. of saturated steam at the valve setpoint. Below 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. There are no credible accidents which could occur when the Reactor Coolant System is connected to the Residual Heat Removal System which could give a surge rate exceeding the capacity of one pressurizer safety valve. Also, two safety valves have a capacity greater than the maximum surge rate resulting from complete loss of load.(2)

The limitation specified in item 4 above on reactor coolant loop isolation will prevent an accidental isolation of all the loops which would eliminate the capability of dissipating core decay heat when the Reactor Coolant System is not connected to the Residual Heat Removal System.

The requirement for steam bubble formation in the pressurizer when the reactor has passed 1% subcriticality will ensure that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that 125 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 at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

References:

(1) FSAR Section 14.2.9

(2) FSAR Section 14.2.10

1. One residual heat removal pump may be out of service, provided immediate attention is directed to making repairs.
2. One residual heat removal heat exchanger may be out of service, provided immediate attention is directed to making repairs.

Basis

The Residual Heat Removal System is required to bring the Reactor Coolant System from conditions of approximately 350°F and pressures between 400 and 450 psig to cold shutdown conditions. Heat removal at greater temperatures is by the Steam and Power Conversion System. The Residual Heat Removal System is provided with two pumps and two heat exchangers. If one of the two pumps and/or one of the two heat exchangers is not operative, safe operation of the unit is not affected; however, the time for cooldown to cold shutdown conditions is extended.

The NRC requires that the series motorized valves in the line connecting the RHRS and RCS be provided with pressure interlocks to prevent them from opening when the reactor coolant system is at pressure.

References

FSAR Section 9.3 - Residual Heat Removal System.

3.6 TURBINE CYCLE

Applicability

Applies to the operating status of the Main Steam and Auxiliary Feed Systems.

Objective

To define the conditions required in the Main Steam System and Auxiliary Feed System for protection of the steam generator and to assure the capability to remove residual heat from the core during a loss of station power.

Specification

A. A unit's Reactor Coolant System temperature or pressure shall not exceed 350°F or 450 psig, respectively, or the reactor shall not be critical unless the five main steam line code safety valves associated with each steam generator in unisolated reactor coolant loops, are operable.

B. To assure residual heat removal capabilities, the following conditions shall be met prior to the commencement of any unit operation that would establish reactor coolant system conditions of 350°F and 450 psig which would preclude operation of the Residual Heat Removal System.

1. Two motor driven auxiliary feedwater pumps shall be operable and one of three auxiliary feedwater pumps for the opposite unit shall be operable.

2. A minimum of 96,000 gal of water shall be available in the tornado missile protected condensate storage tank to supply emergency water to the auxiliary feedwater pump suctions. A minimum of 60,000 gal of water shall be available in the tornado protected condensate storage tank of the opposite unit to supply emergency water to the auxiliary feedwater pump suction of that unit.

3. All main steam line code safety valves, associated with steam generators in unisolated reactor coolant loops, shall be operable.

C. Prior to reactor power exceeding 10%, the steam driven auxiliary feedwater pumps shall be operable.

D. System piping, valves, and control board indication required for the operation of the components enumerated in Specification B. 1, 2, 3, and C shall be operable.

E. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to operable status within 72 hours or be in hot shutdown within the following 12 hours.

F. With two auxiliary feedwater pumps inoperable, be in at least hot shutdown within 6 hours and the reactor coolant system temperature and pressure less than 350°F and 450 psig, respectively, within the following 6 hours.

G. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to operable status as soon as possible.

H. With no operable auxiliary feedwater pump available from the opposite unit, restore one auxiliary feedwater from the opposite unit to operable status within 72 hours or be in hot shutdown within the following 12 hours.

I. The iodine - 131 activity in the secondary side of any steam generator, in an unisolated reactor coolant loop, shall not exceed 9 curies. Also the specific activity of the secondary coolant system shall be $\leqq 0.10$ µCi/cc DOSE EQUIVALENT I-131. If the specific activity of the secondary coolant system exceeds 0.10 µCi/cc DOSE EQUIVALENT I-131, the reactor shall be shutdown and cooled to 500°F or less within 6 hours after detection and in the Cold Shutdown Condition within the following 30 hours.

J. The requirements of Specification B-2 above may be modified to allow utilization of protected condensate storage tank water with the auxiliary steam generator feed pumps provided the water level is maintained above 60,000 gallons, sufficient replenishment water is available in the 300,000 gallon condensate storage tank, and replenishment of the protected condensate storage tanks is commenced within two hours after the cessation of protected condensate storage tank water consumption.

Basis

A reactor which has been shutdown from power requires removal of core residual heat. While reactor coolant temperature or pressure is greater than 350°F or 450 psig, respectively, residual heat removal requirements are normally satisfied by steam bypass to the condenser. If the condenser is unavailable, steam can be released to the atmosphere through the safety valves, power operated relief valves, or the 4 inch decay heat release line.

The capability to supply feedwater to the generators is normally provided by the operation of the Condensate and Feedwater Systems. In the event of complete loss of electrical power to the station, residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps and the 110,000 gallon condensate storage tank. In the event of a fire which would render the auxiliary feedwater pumps inoperable, residual heat removal would continue to be assured by the availability of either the steam driven auxiliary feedwater pump or one of the motor driven auxiliary feedwater pumps from the opposite unit.

A minimum of 92,000 gallons of water in the 110,000 gallon condensate tank is sufficient for 8 hours of residual heat removal following a reactor trip and loss of all off-site electrical power. If the protected condensate storage tank level is reduced to 60,000 gallons, the immediately available replenishment water in the 300,000 gallon condensate tank can be gravity-feed to the protected tank if required for residual heat removal. An alternate supply of feedwater to the auxiliary feedwater pump suctions is also available from the Fire Protection System Main in the auxiliary feedwater pump cubicle.

The five main steam code safety valves associated with each steam generator have a total combined capacity of 3,725,575 pounds per hour at their individual set pressure; the total combined capacity of all fifteen main steam code safety valves is 11,176,725 pounds per hour. The ultimate power rating steam flow is 11,167,923 pounds per hour. The combined capacity of the safety valves required by Specification 3.6 always exceeds the total steam flow corresponding to the maximum steady-state power than can be obtained during one, two, or three reactor coolant loop operation.

The operability of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each motor driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 350 gpm at a pressure of 1080 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1080 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant system temperature to less than 350°F when the Residual Heat Removal System may be placed in operation.

The availability of the auxiliary feedwater pumps, the protected condensate storage tank, and the main steam line safety valves adequately assures that sufficient residual heat removal capability will be available when required.

The limit on steam generator secondary side iodine - 131 activity is based on limiting inhalation thyroid does at the site boundary of 1.5 rem after a postulated accident that would result in the release of the entire contents of a unit's steam generators to the atmosphere. In this accident, with the halogen inventories in the steam generator being at equilibrium valves, I-131 would contribute 75 percent of the resultant thyroid dose at the site boundary; the remaining 25 percent of the dose is from other isotopes of iodine. In the analysis, one-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plate out and retention in water droplets.

The inhalation thyroid dose at the site boundary is given by:

$$\text{Dose (Rem)} = \frac{(C)\ (X/Q)\ (D\infty/A)\ (B.R.)}{(.75)\ (P.F.)}$$

where: C = steam generator I-131 activity (curies)

$X/Q = 8.14 \times 10^{-4}$ sec/m^3

$D\infty/A = 1.48 \times 10^{6}$ rem/Ci for I-131

B.R. = breathing rate, 3.47×10^{-4} m^3/sec. from TID 14844

P.F. = plating factor, 10

Assuming the postulated accident, the resultant thyroid dose is 1.5 rem. The steam generator's specific iodine - 131 activity limit is calculated by dividing the total activity limit of 9 curies by the water volume of a steam generator. A full power, with a steam generator water volume of 47.6 M^3, the specific iodine - 131 limit would be .18 μCi/cc; at zero power, with a steam generator water volume of 101 M^3, the specific iodine - 131 limit would be .089 μCi/cc.

The limitation on secondary system specific activity ensures that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture.

References

FSAR Section 4	Reactor Coolant System
FSAR Section 9.3	Residual Heat Removal System
FSAR Section 10.3.1	Main Steam System
FSAR Section 10.3.2	Auxiliary Steam System
FSAR Section 10.3.5	Auxiliary Feedwater Pumps
FSAR Section 10.3.8	Vent and Drain Systems
FSAR Section 14.3.2.5	Environmental Effects of a Steam Line Break

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability:

Applies to reactor and safety features instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

A. For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with TS Tables 3.7-1 through 3.7-3.

B. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 4 of TS Tables 3.7-1 through 3.7-3.

C. In the event of sub-system instrumentation channel failure permitted by specification 3.7-B, TS Tables 3.7-1 through 3.7-3 need not be observed during the short period of time and operable sub-system channel are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

D. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in TS Table 3.7-4.

E. Automatic functions operated from radiation monitor alarms shall be as stated in TS Table 3.7-5.

F. The accident monitoring instrumentation for its associated operable components listed in TS Table 3.7-6 shall be operable in accordance with the following:

1. With the number of operable accident monitoring instrumentation channels less than the total number of channels shown in TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 7 days or be in at least hot shutdown within the next 12 hours.

2. With the number of operable accident monitoring instrumentation channels less than the minimum channels operable requirements of TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

Basis

Instrument Operating Conditions

During plant operations, the complete instrumentation system will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; and (b) placing the $\Delta T/T_{avg}$ protection channel set that is being fed from the NIS channel in the trip mode and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features [1].

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break Accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safeguards Instrumentation has been designed to sense these effects of the Loss of Coolant accident by detecting low pressurizer pressure to generator signals actuating the SIS active phase. The SIS active phase is also actuated by a high containment pressure signal brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between the steam header and steam generator line or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line break accident is also provided by low pressurizer pressure actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features also initiate containment spray upon sensing a high-high containment pressure signal. The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The containment spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high-high containment pressure sensed by 3 out of the 4 containment pressure signals provided for its actuation.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line trip valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high-high containment pressure or high steam line flow with coincident low steam line pressure or low reactor coolant average temperature. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effects of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steam line break inside the containment, by stopping the entry of feedwater.

Auxiliary Feedwater System Actuation

The automatic initiation of auxiliary feedwater flow to the steam generators by instruments identified in Table 3.7-2 ensures that the Reactor Coolant System Decay Heat can be removed following loss of main feedwater flow. This is consistent with the requirements of the "TMI-2 Lesson Learned Task Force Status Report", NUREG-0578, item 2.1.7.b.

Setting Limits

1. The high containment pressure limit is set at about 10% of design containment pressure. Initiation of Safety Injection protects against loss of coolant[2] or steam line break[3] accidents as discussed in the safety analysis.

2. The high-high containment pressure limit is set at about 50% of design containment pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant[2] or steam line break accidents[3] as discussed in the safety analysis.

3. The pressurizer low pressure setpoint for safety injection actuation is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis.[2]

4. The steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis.[3]

5. The high steam line flow differential pressure setpoint is constant at 40% full flow between no load and 20% load and increasing linearly to 110% of full flow at full load in order to protect against large steam line break accidents. The coincident low T_{avg} setting limit for SIS and steam line isolation initiation is set below its hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.[3]

Automatic Functions Operated from Radiation Monitors

The Process Radiation Monitoring System continuously monitors selected lines containing or possibly containing, radioactive effluent. Certain channels in this system actuate control valves on a high-activity alarm signal. Additional information on the Process Radiation Monitoring System is available in the FSAR.[4]

Accident Monitoring Instrumentation

The operability of the accident monitoring instrumentation in Table 3.7-6 ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. On the pressurizer PORVs, the pertinent channels consist of limit switch indication and acoustic monitor indication. The pressurizer safety valves utilize an acoustic monitor channel and a downstream high temperature indication channel. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations".

References

(1) FSAR - Section 7.5

(2) FSAR - Section 14.5

(3) FSAR - Section 14.3.2

(4) FSAR - Section 11.3.3

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
1.	Manual	1	--		Maintain hot shutdown
2.	Nuclear Flux Power Range	3	2	Low trip setting when 2 of 4 power channels greater than 10% of full power	Maintain hot shutdown
3.	Nuclear Flux Intermediate Range	1	--	2 of 4 power channels greater than 10% full power	Maintain hot shutdown
4.	Nuclear Flux Source Range	1	--	1 of 2 intermediate range channels greater than 10^{-10} amps	Maintain hot shutdown
5.	Overtemperature ΔT	2	1		Maintain hot shutdown
6.	Overpower ΔT	2	1		Maintain hot shutdown
7.	Low Pressurizer Pressure	2	1	3 of 4 nuclear power channels and 2 of 2 turbine load channels less than 10% of rated power	Maintain hot shutdown
8.	Hi Pressurizer Pressure	2	1	Same as Item 7 above	Maintain hot shutdown

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
9.	Pressurizer-Hi Water Level	2	1	3 of 4 nuclear power channels and 2 of 2 turbine load channels less than 10% of rated power	Maintain hot shutdown
10.	Low Flow	2/operable loop		If inoperable loop channels are not in service they must be placed in the tripped mode	Maintain hot shutdown
11.	Turbine Trip	2	1		Maintain less than 10% rated power
12.	Lo-Lo Steam Generator Water Level	2/non-isolated loop	1/non-isolated loop		Maintain hot shutdown
13.	Underfrequency 4KV Bus	2	1		Maintain hot shutdown
14.	Undervoltage 4KV Bus	2	1		Maintain hot shutdown

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	1	2	3	4
FUNCTIONAL UNIT	MIN. OPERABLE CHANNELS	DEGREE OF REDUN-DANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
15. Control rod misalignment Monitor**				
a) rod position deviation	1	-		Log individual rod positions once/hour, and after a load change > 10% or after > 30 inches of control rod motion.
b) quadrant power tilt monitor (upper and lower excore neutron detectors)	1	-		Log individual upper upper and lower ion chamber currents once/hour and after a load change > 10% or after > 30 inches of control rod motion
16. Safety Injection	See Item 1 of TS Table 3.7-2			

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	1	2	3	4
FUNCTIONAL UNIT	MIN. OPERABLE CHANNELS	DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
17. Low steam generator water level with steam/feedwater mismatch flow	1/non-isolated loop 1/non-isolated loop	-- --		Maintain hot shutdown

**If both rod misalignment monitors (a and b) inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93 percent of rated power in addition to the increased surveillance noted.

TABLE 3.7-2
ENGINEERED SAFEGUARDS ACTION

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 MIN. DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
	1. SAFETY INJECTION				
a.	Manual	1	0		Cold shutdown
b.	High Containment Press. (Hi Setpoint)	3	1		Cold shutdown
c.	High Differential Press. between any Steam Line and the Steam Line Header	2/non-isolated loop	1/non-isolated loop		Cold shutdown
d.	Pressurizer Low-Low Press.	2	1	Primary Pressure less than 2000 psig except when reactor is critical	Cold shutdown
e.	High Steam Flow in 2/3 Steam Lines with Low T_{avg} or Low Steam Line Press.	1/steamline 2 T_{avg} signals 2 Steam Press. Signals	*** 1 1	Reactor Coolant average temperature less than 547°F during heatup and cooldown	Cold shutdown

TABLE 3.7-2
ENGINEERED SAFEGUARDS ACTION

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 MIN. DEGREE OF REDUN-DANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
	2. CONTAINMENT SPRAY				
a.	Manual	2	**		Cold shutdown
b.	High Containment Press. (Hi-Hi Setpoint)	3	1		Cold shutdown
	3. AUXILIARY FEEDWATER				
a.	Steam Generator Water Level Low-Low				
	i. Start Motor Driven Pumps	2/Stm. Gen.	1	Loop Stop Valve in res-pective loop closed	Place inoperable channel in Tripped condition within one hour
	ii. Start Turbine Driven Pumps	2/Stm. Gen	1		
b.	RCP Undervoltage Start Turbine Driven Pump	2	1		Place inoperable channel in Tripped condition within one hour
c.	Safety Injection Start Motor Driven Pumps	(All safety injection initiating functions and requirements)			
d.	Station Blackout Start Motor Driven Pump	2	0		Restore inoperable channel within 48 hours or be in hot shutdown within next 6 hours.

TABLE 3.7-2
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 MIN. DEGREE OF REDUN-DANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
e.	Trip of Main Feedwater Pumps Start Motor Pumps	1/Pump	1/Pump		Restore inoperable channel within 48 hours or be in hot shutdown within next 6 hours.

** Must actuate 2 switches simultaneously

*** With specified minimum operable channels the 2/3 high steam flow is already in the trip mode

TABLE 3.7-3
INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUN-DANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
1.	CONTAINMENT ISOLATION				
a.	Safety Injection	See Item No. 1 of Table 3.7-2			Cold shutdown
b.	Manual	1	--		Hot shutdown
c.	High Containment Press. (Hi setpoint)	3	1		Cold shutdown
d.	High Containment Press.	3	1		Cold shutdown
2.	STEAM LINE ISOLATION				
a.	High Steam Flow in 2/3 lines and 2/3 Low T_{avg} or 2/3 Low Steam Pressure	1/steamline 2/T_{avg} signals 2 Stm. Press. signals	*** 1 1		Cold shutdown
b.	High Containment Press. (Hi-Hi Level)	3	1		Cold shutdown
c.	Manual	1/line	--		Hot shutdown
3.	FEEDWATER LINE ISOLATION				
a.	Safety Injection	See Item No. 1 of Table 3.7-2			Cold shutdown

*** With the specified minimum operable channels the 2/3 high steam flow is already in the trip mode.

TABLE 3.7-4
ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1	High Containment Pressure (High Containment Pressure Signal)	a) Safety Injection b) Containment Vacuum Pump Trip c) High Press. Containment Iso. d) Safety Injecton Contain. Iso. e) F.W. Line Isolation	≦5 psig
2	High High Containment Pressure (High High Containment Pressure Signals)	a) Containment Spray b) Recirculation Spray c) Steam Line Isolation d) High High Press. Contain. Iso.	≦25 psig
3	Pressurizer Low Low Pressure	a) Safety Injection b) Safety Injection Contain. Iso. c) Feedwater Line Isolation	≧1,700 psig
4	High Differential Pressure Between Steam Line and the Steam Line Header	a) Safety Injection b) Safety Injection Contain. Iso. c) F.W. Line Isolation	≦150 psi
5	High Steam Flow in 2/3 Steam Lines	a) Safety Injection	≦40% (at zero load) of full steam flow ≦40% (at 20% load) of full steam flow
		b) Steam Line Isolation c) Safety Injection Contain. Iso. d) F.W. Line Isolation	≦110% (at full load) of full steam flow
	Coincident with Low T_{avg} or Low Steam		≧541°F T_{avg}
	Line Pressure		≧500 psig steam line pressure

TABLE 3.7-4
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NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6	AUXILIARY FEEDWATER		
a.	Steam Generator Water Level Low-Low	Aux. Feedwater Initiation S/G Blowdown Isolation	≥5% narrow range
b.	RCP Undervoltage	Aux. Feedwater Initiation	≥70% nominal
c.	Safety Injection	Aux. Feedwater Initiation	All S.I. setpoints
d.	Station Blackout	Aux. Feedwater Initiation	≥46.7% nominal
e.	Main Feedwater Pump Trip	Aux. Feedwater Initiation	N.A.

TABLE 3.7-5

AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

	MONITOR CHANNEL	AUTOMATIC FUNCTION AT ALARM CONDITIONS	MONITORING REQUIREMENTS	ALARM SETPOINT µCI/cc
1.	Process vent particulate and gas monitors (RM-GW-101 & RM-GW-102)	Stops discharge from contain. vacuum systems and waste gas decay tanks (shuts, Valve Nos. RCV-GW-160, FCV-GW-260, FCV-GW-101)	See Specifications 3.11 and 4.9	Particulate $\leq 4\times10^{-8}$ Gas $\leq 9\times10^{-2}$
2.	Component cooling water radiaton monitors (RM-CC-105 & RM-CC-106)	Shuts surge tank vent valve HCV-CC-100	See Specifications 3.13 and 4.9	≤Twice Background
3.	Liquid waste disposal radiation monitors (RM-LW-108)	Shuts effluent discharge valves FCV-LW-104A and FCV-LW-104B	See Specifications 3.11 and 4.9	$\leq 1.5\times10^{-3}$
4.	Condenser air ejector radiation monitors (RM-SV-111 & RM-SV-211)	Diverts flow to the containment of the affected unit (Opens TV-SV-102 and shuts TV-SV-103 or opens TV-SV-202 and shuts TV-SV-203)	See Specifications 3.11 and 4.9	≤1.3
5.	Containment particulate and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260)	Trips affected unit's purge supply and exhaust fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specifications 3.10 and 4.0	Particulate $\leq 9\times10^{-9}$ Gas $\leq 1\times10^{-5}$
6.	Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262)	Trips affected unit's purge supply and exhaust fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D	See Specifications 3.10 and 4.9	≤50 mrem/hr

TABLE 3.7-6
ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Auxiliary Feedwater Flow Rate	1 per S/G	1 per S/G
2.	Reactor Coolant System Subcooling Margin Monitor	2	1
3.	PORV Position Indicators	2/valve	1/valve
4.	PORV Block Valve Position Indicator	1/valve	1/valve
5.	Safety Valve Position Indicators	2/valve	1/valve

3.10 REFUELING

Applicability

Applies to operating limitations during refueling operations.

Ojective

To assure that no accident could occur during refueling operations that would affect public health and safety.

Specification

A. During refueling operations the following conditions are satisfied:

1. The equipment door and at least one door in the personnel air lock shall be properly closed. For those systems which provide a direct path from containment atmosphere to the outside atmosphere, all automatic containment isolation valves in the unit shall be operable or at least one valve shall be closed in each line penetrating the containment.

2. The Containment Vent and Purge System and the area and airborne radiation monitors which initiate isolation of this system, shall be tested and verified to be operable immediately prior to refueling operations.

3. At least one source range neutron detector shall be in service at all times when the reactor vessel head is unbolted. Whenever core geometry or coolant chemistry is being changed, subcritical neutron flux shall be continuously monitored by at least two source range neutron detectors, each with continuous visual indication in the Main Control Room and one with audible indication within the containment. During core fuel loading phases, there shall be a minimum neutron count rate detectable on two operating source range neutron detectors with the exception of initial core loading, at which time a minimum neutron count rate need be established only when there are eight (8) or more fuel assemblies loaded into the reactor vessel.

4. Manipulator crane area radiation levels and airborne activity levels within the containment and airborne activity levels in the ventilation exhaust duct shall be continuously monitored during refueling. A manipulator crane high radiation alarm or high airborne activity level alarm within the containment will automatically stop the purge ventilation fans and automatically close the containment purge isolation valves.

5. Fuel pit bridge area radiation levels and ventilation vent exhaust airborne activity levels shall be continuously monitored during refueling. The fuel building exhaust will be continuously bypassed through the iodine filter bank during refueling procedures, prior to discharge through the ventilation vent.

6. At least one residual heat removal pump and heat exchanger shall be operable to circulate reactor coolant. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations or reactor vessel surveillance inspections.

7. Two residual heat removal pumps and heat exchangers shall be operable to circulate reactor coolant when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

8. At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange during movement of fuel assemblies.

9. When the reactor vessel head is unbolted, a minimum boron concentration of 2,000 ppm shall be maintained in any filled portion of the Reactor Coolant System and shall be checked by sampling at least once every 8 hours.

10. Direct communication between the Main Control Room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

11. No movement of irradiated fuel in the reactor core shall be accomplished until the reactor has been subcritical for a period of at least 100 hours.

12. A spent fuel cask or heavy loads exceeding 110 percent of the weight of a fuel assembly (not including fuel handling tool) shall not be moved over spent fuel, and only one spent fuel assembly will be handled at one time over the reactor or the spent fuel pit.

13. A spent fuel cask shall not be moved into the Fuel Building until such time as the NRC has reviewed and approved the spent fuel cask drop evaluation.

B. If any one of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the conditions so that the specified limit is met, and no operations which increase the reactivity of the core shall be made.

C. After initial fuel loading and after each core refueling operation and prior to reactor operation at greater than 75% of rated power, the movable incore detector system shall be utilized to verify proper power distribution.

Basis

Detailed instructions, the above specified precautions and the design of the fuel handling equipment, which incorporates built-in interlocks and safety features, provide assurance that an accident, which would result in a hazard to public health and safety, will not occur during refueling operations. When no change is being made in core geometry, one neutron detector is sufficient to monitor the core and permits maintenance of the out-of-function instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. Containment high radiation levels and high airborne activity levels automatically stop and isolate the Containment Purge System. The fuel building ventilation exhaust is diverted through charcoal filters whenever refueling is in progress. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

The shutdown margin established by Specification A-8 maintains the core subcritical, even with all of the control rod assemblies withdrawn from the core. During refueling, the reactor refueling water cavity is filled with approximately 220,000 gal of water borated to at least 2,000 ppm boron. The boron concentration of this water is sufficient to maintain the reactor subcritical by approximately 10% Δ k/k in the cold shutdown condition with all control rod assemblies inserted and also to maintain the core subcritical by approximately 1% with no control rod assemblies inserted into the reactor. Periodic checks of refueling water boron concentration assure the proper shutdown margin. Specification A-9 allows the Control Room Operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are used during refueling to assure safe handling of the fuel assemblies. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accomodate only one fuel assembly at a time.

Upon each completion of core loading and installation of the reactor vessel head, specific mechanical and electrical tests will be performed prior to initial criticality.

The fuel handling accident has been analyzed based on the activity that could be released from fuel rod gaps of 204 rods of the highest power assembly* with a 100 hour decay period following power operation at 2550 MWt for 23,000 hours. The requirements detailed in Specification 3.10 provide assurance that refueling unit conditions conform to the operating conditions assumed in the accident analysis.

Detailed procedures and checks insure that fuel assemblies are loaded in the proper locations in the core. As an additional check, the moveable incore detector system will be used to verify proper power distribution. This system is capable of revealing any assembly enrichment error or loading error which could cause power shapes to be peaked in excess of design value.

* Fuel rod gap activity from 204 rods of the highest power 15x15 assembly is greater than fuel rod gap activity from 264 rods of the highest power 17x17 demonstration assembly.

References

FSAR Section 5.2	Containment Isolation
FSAR Section 6.3	Consequence Limiting Safeguards
FSAR Section 9.12	Fuel Handling System
FSAR Section 11.3	Radiation Protection
FSAR Section 13.3	Table 13.3-1
FSAR Section 14.4.1	Fuel Handling Accidents
FSAR Supplement: Volume I:	Question 3.2

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

A. Calibration, testing, and checking of instrumentation channels shall be performed as detailed in Table 4.1-1.

B. Equipment tests shall be conducted as detailed below and in Table 4.1-2A.

1. Each Pressurizer PORV shall be demonstrated operable:
    a. At least once per 31 days by performance of a channel functional test, excluding valve operation, and
    b. At least once per 18 months by performance of a channel calibration.

2. Each Pressurizer PORV block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel.

3. The pressurizer water volume shall be determined to be within its limit as defined in Specification 2.3.A.3.a at least once per 12 hours whenever the reactor is not subcritical by at least 1% $\Delta k/k$.

C. Sampling tests shall be conducted as detailed in Table 4.1-2B.

D. Whenever containment integrity is not required, only the asterisked items in Table 4.1-1 and 4.1-2A and 4.1-2B are applicable.

E. Flushing of sensitized stainless steel pipe sections shall be conducted as detailed in TS Table 4.1-3A and 4.1-3B.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
1.	Nuclear Power Range	S M(3)	D (1) Q (3)	BW(2)	1) Against a heat balance standard 2) Signal of ΔT; bistable action (permissive, rod stop, strips) 3) Upper and lower chambers for symetric offset by means of the moveable incore detector system.
2.	Nuclear Intermediate Range	*S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	*S(1)	N.A.	P(2)	1) Once/Shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	*S	R	BW(1) BW(2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure (High & Low)	S	R	M	
8.	4 Kv Voltage and Frequency	S	R	M	Reactor protection circuit only
9.	Analog Rod Position	*S(1,2) (4)	R	M(3)	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) Rod bottom bistable action 4) NA when reactor is in cold shutdown

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
10.	Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15A.	Unit 1 Refueling Water Storage Tank Level	W	R	N.A.	
15B.	Unit 2 Refueling Water Storage Tank Level	S	R	M	
16.	Boron Injection Tank Level	W	N.A.	N.A.	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Reactor Containment Pressure-CLS	*D	R	M(1)	1) Isolation Valve signal and spray signal
19.	Processing and Area Radiation Monitoring Systems	*D	R	M	
20.	Boric Acid Control	N.A.	R	N.A.	
21.	Containment Sump Level	N.A.	R	N.A.	
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Containment Pressure-Vacuum Pump System	S	R	N.A.	
24.	Steam Line Pressure	S	R	M	

TABLE 4.1-1

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
25.	Turbine First Stage Pressure	S	R	M	
26.	Emergency Plan Radiation Instr.	*M	R	M	
27.	Environmental Radiation Monitors	*M	N.A.	N.A.	TLD Dosimeters
28.	Logic Channel Testing	N.A.	N.A.	M	
29.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	R	
30.	Turbine Trip Setpoint	N.A.	R	R	Stop valve closure or low EH fluid pressure
31.	Seismic Instrumentation	M	SA	M	
32.	Reactor Trip Breaker	N.A.	N.A.	M	
33.	Reactor Coolant Pressure (Low)	N.A.	R	N.A.	
34.	Auxiliary Feedwater				
	a. Steam Generator Water Level Low-Low	S	R	M	
	b. RCP Undervoltage	N.A.	R	N.A.	
	c. S.I.	(All Safety Injection surveillance requirements)			
	d. Station Blackout	N.A.	R	N.A.	
	e. Main Feedwater Pump Trip	N.A.	N.A.	R	

S - Each shift　　M - Monthly
D - Daily　　P - Prior to each startup if not done previous week
W - Weekly　　R - Each Refueling Shutdown
NA - Not applicable　　BW - Every two weeks
SA - Semiannually　　AP - After each startup if not done previous week
Q - Every 90 effective full power days

* See Specification 4.1D

TABLE 4.1-2

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Auxiliary Feedwater Flow Rate	N.A.	R
2.	Reactor Coolant System Subcooling Margin Monitor	M	R
3.	PORV Position Indicators	M	R
4.	PORV Block Valve Position Indicator	N.A.	R
5.	Safety Valve Position Indicators	M	R

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1.	Control Rod Assemblies	Rod drop times of all full length rods at hot and cold conditions	Each refueling shutdown or after disassembly or maintenance requiring the breech of the Reactor Coolant System integrity	7
2.	Control Room Assemblies	Partial movement of all rods	Every 2 weeks	7
3.	Refueling Water Chemical Addition Tank	Functional	Each refueling shutdown	6
4.	Pressurizer Safety Valves	Setpoint	Each refueling shutdown	4
5.	Main Steam Safety Valves	Setpoint	Each refueling shutdown	10
6.	Containment Isolation Trip	*Functional	Each refueling shutdown	5
7.	Refueling System Interlocks	*Functional	Prior to refueling	9.12
8.	Service Water System	*Functional	Each refueling shutdown	9.9
9.	Fire Protection Pump and Power Supply	Functional	Monthly	9.10
10.	Primary System Leakage	*Evaluate	Daily	4
11.	Diesel Fuel Supply	*Fuel Inventory	5 days/week	8.5
12.	Boric Acid Piping Heat Tracing Circuits	*Operational	Monthly	9.1
13.	Main Steam Line Trip	Functional (1) Full closure (2) Partial closure	 (1) Each cold shutdown (2) Before each startup	10

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
14.	Service Water System Valves in Line Supplying Recirculation Spray Heat Exchangers	Functional	Each refueling	9.9
15.	Control Room Ventilation System	*Ability to maintain positive pressure for 1 hour using a volume of air equivalent to or less than stored in the bottled air supply	Each refueling interval (approx. every 12-18 months)	9.13
16.	Reactor Vessel Overpressure Mitigating System (except backup air supply)	Functional & Setpoint	Prior to decreasing RCS temperature below 350°F and monthly while the RCS is <350°F and the Reactor Vessel Head is bolted	None
17.	Reactor Vessel Overpressure Mitigating System Backup Air Supply	Setpoint	Refueling	None

* See Specification 4.1.D

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, SAFETY AND OPERATION REVIEW

Specification

A. The Station Manager shall be responsible for the safe operation of the facility. In his absence, the Assistant Station Manager shall be responsible for the safe operation of the facility. During the absence of both, the Station Manager shall delegate in writing the succession to this responsibility.

1. The offsite organization for facility management and technical support shall be as shown on TS Figure 6.1-1.

B. The Station organization shall conform to the chart as shown on TS Figure 6.1-2.

1. Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N.18.1-1971 for comparable positions, and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, except for the Supervisor-Health Physics who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

2. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents. The requirement for the Shift Technical Advisor becomes effective on January 1, 1981.

3. The Station Manager is responsible for ensuring that retraining and replacement training programs for the facility staff are maintained and that such programs meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the SEC staff.

4. Each on duty shift shall be composed of at least the minimum shift crew composition for each unit as shown in Table 6.1-1.

5. A health physics technician shall be on site when fuel is in the reactor.

6. A fire team of at least five members, all of whom have received fire service training, will be maintained on-site at all times. This excludes personnel in Table 6.1-1 of the minimum shift crew necessary for safe shutdown of the plant and any personnel requried for other essential functions during a fire emergency.

7. A training program for the fire brigade and fire teams shall be maintained under the directions of a Fire Marshall and shall meet or exceed the requirements of the NFPA Code Section 27 (1975), except that training sessions and drills shall be held at least once per 92 days.

8. The health physics technician and Fire Brigade composition of Specifications 6.1.B.5 and 6.1.B.6 may be less than the minimum requirement for a period of time not to exceed 2 hours in order to accomodate unexpected absence provided immediate action is taken to fill the required positions.

TABLE 6.1-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	ONE UNIT OPERATING	TWO UNITS OPERATING	TWO UNITS IN COLD SHUTDOWN OR REFUELING
SS	1	1	1
SRO	1	1	None
RO	3	3	2
AO	3	3	3
STA	1	1	None

TABLE 6.1-1 (Continued)

SS	-	Shift Supervisor with a Senior Reactor Operators License.
SRO	-	Individual with a Senior Reactor Operators License.
RO	-	Individual with a Reactor Operators License.
AO	-	Auxiliary Operator
STA	-	Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.1-1 for a period of time not to exceed 2 hours in order to accomodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.1-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in operation, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is shutdown or refueling, an individual with a valid RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command functions.

H. Practice of site evacuation exercises shall be conducted annually, following emergency procedures and including a check of communications with off-site report groups. An annual review of the Emergency Plan will be performed.

I. The industrial security program which has been established for the station shall be implemented, and appropriate investigation and/or corrective action shall be taken if the provisions of the program are violated. An annual review of the program shall be performed.

J. The facility fire protection program and implementing procedures which have been established for the station shall be implemented. The program shall be reviewed at least once every two years.

K <u>Systems Integrity</u>

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

L. <u>Iodine Monitoring</u>

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital area under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.