ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGES

FOR SEQUOYAH NUCLEAR PLANT UNIT 1

1.0 DEFINITIONS

DEFINED TERMS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detectors.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.11 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.12 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.13 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

OFFSITE DOSE CALCULATION MANUAL

1.15 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints. It shall also contain the radiological environmental monitoring program.

OPERABLE - OPERABILITY

1.16 A system, subsystem, train, or component or device shall be OPERABLE or have OPERAILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.17 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.18 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.19 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.20 The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE - PURGING

1.21 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.22 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.23 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.24 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE OCCURRENCE

1.25 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.12 and 6.1.1.13.

SHIELD BUILDING INTEGRITY

1.26 SHIELD BUILDING INTEGRITY shall exist when:

a. The door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

b. The emergency gas treatment system is OPERABLE.

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SOLIDIFICATION

1.28 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a uniformly distributed, monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.28 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.29 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.31 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.32 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

VENTILATION EXHAUST TREATMENT SYSTEM

1.33 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

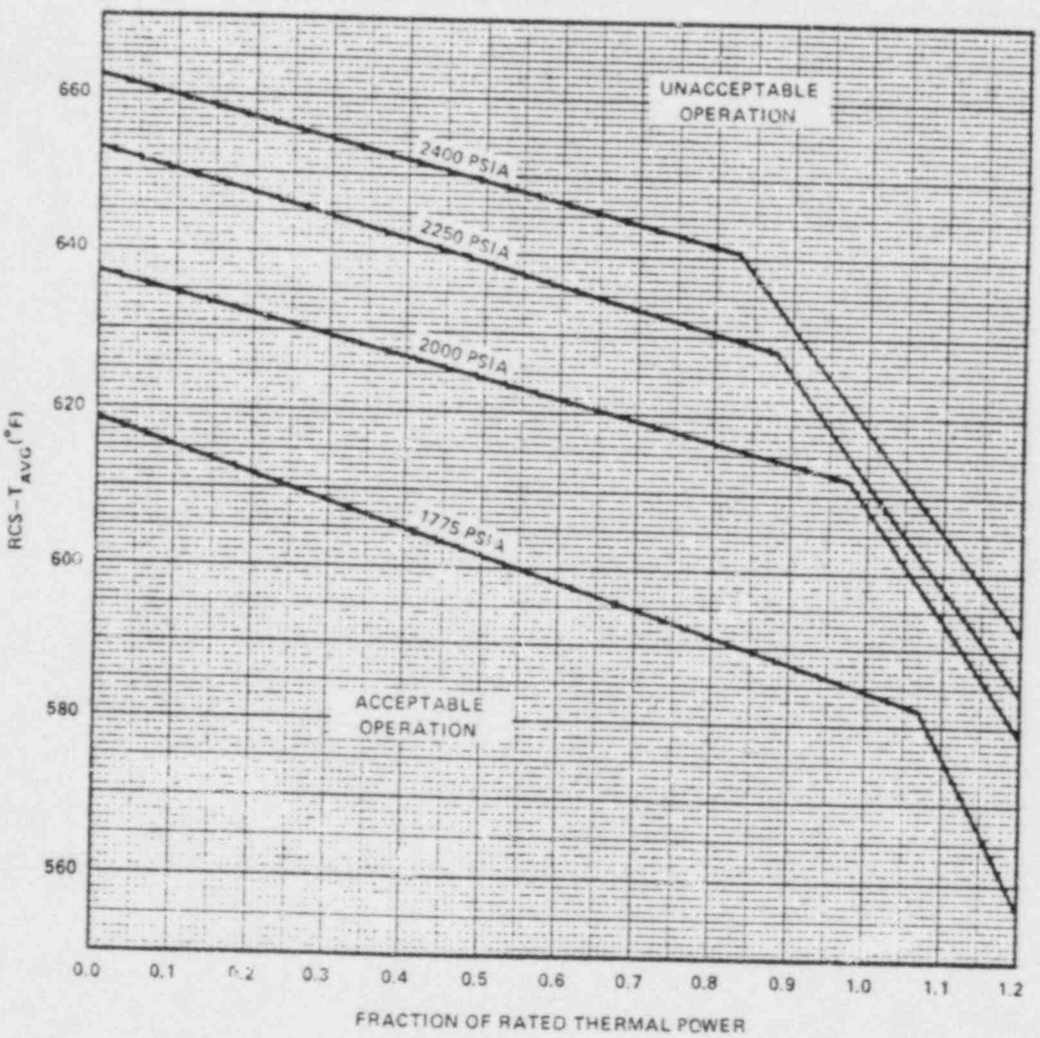

Figure 2.1-1. Reactor Core Safety Limit—Four Loops in Operation

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation and interlocks setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation or interlock setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
21. Turbine Impulse Chamber Pressure - (P-13) Input to Low Power Reactor Trips Block P-7	≤ 10% Turbine Impulse Pressure Equivalent	≤ 11% Turbine Impulse Pressure Equivalent
22. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Trip	≤ 35% of RATED THERMAL POWER	≤ 36% of RATED THERMAL POWER
23. Power Range Neutron Flux - (P-10) - Enable block of Source, Intermediate, and Power Range (low setpoint) reactor Trips	≥ 10% of RATED THERMAL POWER	≥ 9% of RATED THERMAL POWER
24. Reactor Trip P-4	Not Applicable	Not Applicable
25. Power Range Neutron Flux - (P-9) - Blocks Reactor Trip for Turbine Trip Below 50% Rated Power	≤ 50% of RATED THERMAL POWER	≤ 51% of RATED THERMAL POWER

NOTATION

NOTE 1: Overtemperature $\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \left(\frac{1 + \tau_2 S}{1 + \tau_3 S}\right)\left[T\left(\frac{1}{1 + \tau_4 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$

where: $\frac{1}{1 + \tau_1}$ = Lag compensator on measured ΔT

τ_1 = Time constants utilized in the lag compensator for $\Delta T_3 \tau_1$ = 2 secs.

ΔT_o = Indicated ΔT at RATED THERMAL POWER

K_1 $\leq$ 1.14

K_2 = 0.009

SHUTDOWN MARGIN - T_{avg} Less Than or Equal to 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,00 ppm or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Reactor coolant system boron concentration,

   2. Control rod position,

   3. Reactor coolant system average temperature,

   4. Fuel burnup based on gross thermal energy generation,

   5. Xenon concentration, and

   6. Samarium concentration.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System.

b. Two flow paths from the refueling water storage tank via charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is greater than or equal to 145°F when it is a required water source.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

d. At least once per 18 months by verifying that the flow path required by specification 3.1.2.2a delivers at least 10 gpm to the Reactor Coolant System.

ACTION: (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ± 5% target band and ACTION a.2.a)1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% of rated THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 5% target band when at least 2 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ± 5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.14.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive.

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

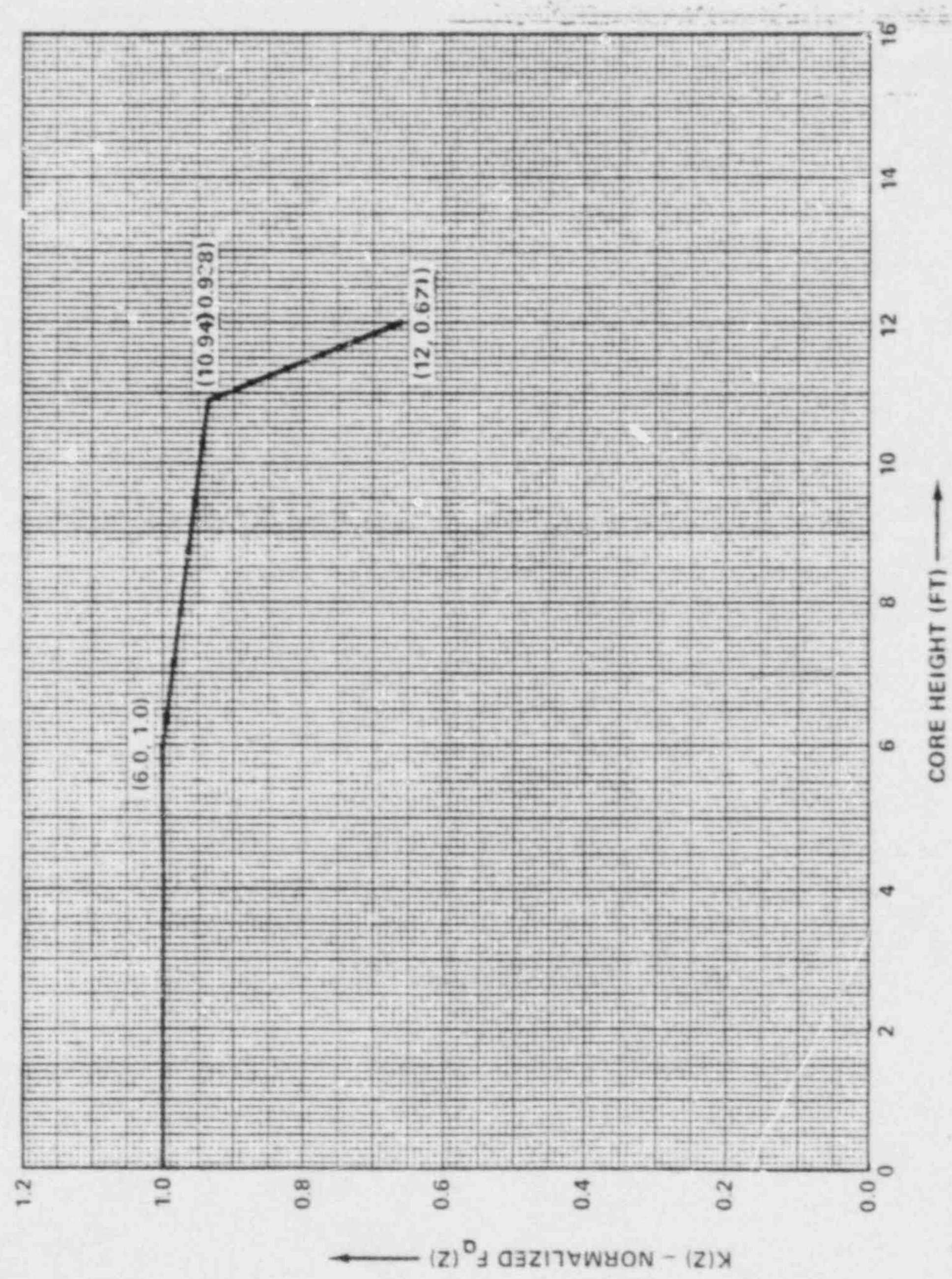

FIGURE 3.2-2.
K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

3/4.2.3 RCS FLOWRATE AN' R

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R_1, R_2 shall be maintained within the regions of allowable operation shown on Figure 3.2-3 for 4 loop operation:

Where:

a. $R_1 = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $R_2 = \dfrac{R_1}{[1 - RBP\ (Bu)]}$,

c. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$,

d. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 3.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$

e. RBP (Bu) = Rod Bow Penalty as a function of region average burnup as shown in Figure 3.2-4, where a region is defined as those assemblies with the same loading date (reloads) or enrichment (first core), and

APPLICABILITY: MODE 1

ACTION:

With the combination of RCS total flow rate and R_1, R_2 outside the regions of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours:

1. Either restore the combination of RCS total flow rate and R_1, R_2 to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High trip setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel and interlock shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceeding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2, and *	1
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2, and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature Delta T					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	4	1**	3	1, 2	9
8. Overpower Delta T					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	4	1**	3	1, 2	9
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Safety Injection Input from ESF	2	1	2	1, 2	12
20. Reactor Trip Breakers	2	1	2	1, 2, and *	12
21. Automatic Trip Logic	2	1	2	1, 2, and *	12
22. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux P-6	2	1	2	2, and*	8a
B. Power Range Neutron Flux - P-7	4	2	3	1	8b
C. Power Range Neutron Flux - P-8	4	2	3	1	8c
D. Power Range Neutron Flux - P-10	4	2	3	1, 2	8d
E. Turbine Impulse Chamber Pressure - P-13	2	1	2	1	8b
F. Power Range Neutron Flux - P-9	4	2	3	1	8e
G. Reactor Trip - P-4	2	1	2	1, 2, and*	14

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal, and fuel in the reactor vessel.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above the P-6 (Block of Source Range Reactor Trip) setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL and the Power Range, Neutron Flux high trip reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors is verified consistent with the normalized symmetric power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, declare the interlock inoperable and verify that all affected channels of the functions listed below are OPERABLE or apply the appropriate ACTION statement(s) for those functions. Functions to be evaluated are:

a. Source Range Reactor Trip.

b. Reactor Trip

Low Reactor Coolant Loop Flow (2 loops)
Undervoltage
Underfrequency
Pressurizer Low Pressure
Pressurizer High Level

c. Reactor Trip

Low Reactor Coolant Loop Flow (1 loop)

d. Reactor Trip

Intermediate Range
Low Power Range
Source Range

e. Reactor Trip

Turbine Trip

ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below the P-8 (Block Low Reactor Coolant Pump Flow) setpoint breaker within the next 2 hours. Operation below the P-8 (Block of Low Reactor Coolant Pump Flow) setpoint breaker may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
14. Main Steam Generator Water Level--Low-Low	≤ 2.0 seconds
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	NOT APPLICABLE
16. Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Trip Breakers	NOT APPLICABLE
21. Automatic Trip Logic	NOT APPLICABLE
22. Reactor Trip System Interlocks	NOT APPLICABLE

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	1, 2, and *
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M	1, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S(7)	R(6)	M and S/U(1)	2, 3, 4, 5, and *
7. Overtemperature Delta T	S	R	M	1, 2
8. Overpower Delta T	S	R	M	1, 2
9. Pressurizer Pressure--Low	S	R	M	1, 2
10. Pressurizer Pressure--High	S	R	M	1, 2
11. Pressurizer Water Level--High	S	R	M	1, 2
12. Loss of Flow - Single Loop	S	R	M	1
13. Loss of Flow - Two Loops	S	R	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
14. Main Steam Generator Water Level--Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Trip Breaker	N.A.	N.A.	M(5) and S/U(1)	1, 2, and *
21. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2, and *
22. Reactor Trip System Interlocks				
A. Intermediate Range Neutron Flux, P-6	N.A	R	S/U (8)	2, and *
B. Power Range Neutron Flux, P-7	N.A.	R	S/U (8)	1
C. Power Range Neutron Flux, P-8	N.A.	R	S/U (8)	1
D. Power Range Neutron Flux, P-10	N.A.	R	S/U (8)	1, 2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	S/U (8)	1
F. Power Range Neutron Flux, P-9	N.A.	R	S/U (8)	1
G. Reactor Trip, P-4	N.A.	R	S/U (8)	1, 2, and *

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Compare incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference greater than or equal to 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Logic only, each startup or when required with the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal if not performed in previous 92 days.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
c. Containment Ventilation Isolation					
1) Manual	2	1	2	1, 2, 3, 4	19
2) Automatic Isclation Logic	2	1	2	1, 2, 3, 4	15
3) Containment Gas Monitor Radioactivity-High	2	1	1	1, 2, 3, 4	19
4) Containmen. Purge Air Exhaust Monitor Radioactivity-High	2	1	1	1, 2, 3, 4	19
5) Containment Particulate Activity High	2	1	1	1, 2, 3, 4	19
4. STEAM LINE ISOLATION					
a. Manual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	25
b. Automatic Actuation Logic	2	1	2	1, 2, 3	23
c. Containment Pressure--High-High	4	2	3	1, 2, 3	18
d. Steam Flow in Two Steam Lines--High				1, 2, 3	
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		16*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line		17

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. AUXILIARY FEEDWATER					
a. Manual Initiation	2	1	2	1, 2, 3	24
b. Automatic Actuation Logic	2	1	2	1, 2, 3	23
c. Main Stm. Gen. Water Level-Low-Low					
i. Start Motor Driven Pumps	3/stm. gen.	2/stm. gen. any stm gen.	2/stm. gen.	1, 2, 3	16
ii. Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen	1, 2, 3	16
d. S.I. Start Motor-Driven Pumps and Turbine Driven Pump	See 1 above (all S.I. initiating functions and requirements)				
e. Station Blackout Start Motor-Driven Pump associated with the shutdown board and Turbine Driven Pump	2/shutdown board	1/shutdown board	2/shutdown board	1, 2, 3	20
f. Trip of Main Feedwater Pumps Start Motor-Driven Pumps and Turbine Driven Pump	1/pump	1/pump	1/pump	1, 2	20*
g. Auxiliary Feedwater Suction Pressure-Low	3/pump	2/pump	2/pump	1, 2, 3	20*

TABLE 3.3-3 (Continued)

ACTION 21 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 22 - With less than the Minimum Number of Channels OPERABLE, declare the interlock inoperable and verify that all affected channels of the functions listed below are OPERABLE or apply the appropriate ACTION statement(s) for those functions. Functions to be evaluated are:

a. Safety Injection
   Pressurizer Pressure

b. Safety Injection
   High St m Line Flow
   Steam Line Isolation
   High Steam Line Flow
   Steam Dump

c. Turbine Trip
   Steam Generator Level High-High
   Feedwater Isolation
   Steam Generator Level High-High

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.

ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 25 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. CONTAINMENT SPRAY		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	$\leq$ 2.81 psig	$\leq$ 2.97 psig
3. CONTAINMENT ISOLATION		
a. Phase "A" Isolation		
1. Manual	Not Applicable	Not Applicable
2. From Safety Injection Automatic Actuation logic	Not Applicable	Not Applicable
b. Phase "B" Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Actuation Logic	Not Applicable	Not Applicable
3. Containment Pressure--High-High	$\leq$ 2.81 psig	$\leq$ 2.97 psig
c. Containment Ventilation Isolation		
1. Manual	Not Applicable	Not Applicable
2. Automatic Isolation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. AUXILIARY FEEDWATER		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Main Steam Generator Water Level-low-low	$\geq$ 21% of narrow range instrument span each steam generator	$\geq$ 20% of narrow range instrument span each steam generator
d. S.I.	See 1 above (all SI Setpoints)	
e. Station Blackout	0 volts with a 5.0 second time delay	0 volts with a 5.0 ± 1.0 second time delay
f. Trip of Main Feedwater Pumps	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low	$\geq$ 2 psig (motor driven pump) $\geq$ 6.5 psig (turbine driven pump)	$\geq$ 1 psig (motor driven pump) $\geq$ 5.5 (turbine driven pump)
7. LOSS OF POWER		
a. 6.9 kv Shutdown Board Undervoltage		
1. Loss of Voltage	0 volts with a 1.5 second time delay	0 volts with a 1.5 ± 0.5 second time delay
2. Load Shedding	0 volts with a 5.0 second time delay	0 volts with a 5.0 ± 1.0 second time delay
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure Manual Block of Safety Injection P-11	$\leq$ 1970 psig	$\leq$ 1980 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS (Continued)		
b. T_{avg} Prevents Manual Block of Safety Injection P-12	$\leq$ 540°F	$\leq$ 542°F
c. T_{avg} Manual Block of Safety Injection, Steam Line Isolation, Block Steam Dump	$\geq$ 540°F	$\geq$ 538°F
d. Steam Generator Level Turbine Trip, Feedwater Isolation P-14	(See 5. above)	
9. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP		
a. RWST Level - Low	130" from tank base	130" ± 4" from tank base
COINCIDENT WITH		
Containment Sump Level - High	30" above elev. 680'	30" ± 2.5" above elev. 680'
AND		
Safety Injection	(See 1 above for all Safety Injection Setpoints/Allowable Valves)	

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0(7)/23.0(1)
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0(2)
d. Containment Isolation-Phase "A"(3)	$\leq$ 18.0(8)/28.0(9)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Essential Raw Cooling Water System	$\leq$ 65.0(8)/75.0(9)
h. Steam Line Isolation	$\leq$ 8.0
i. Emergency Gas Treatment System	$\leq$ 38.0(9)
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 58.00(9)
b. Containment Isolation-Phase "B"	$\leq$ 65(8)/75(9)
c. Steam Line Isolation	$\leq$ 7.0
d. Containment Air Return Fan	$\geq$ 540.0 and $\leq$ 660
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0(2)
9. Main Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M	1, 2, 3
c. Containment Ventilation Isolation				
1) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
2) Automatic Isolation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Gas Monitor Radioactivity-High	S	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
c. Main Steam Generator Water Level-Low-Low	S	R	M	1, 2, 3
d. S.I.	See 1 above (all SI surveillance requirements)			
e. Station Blackout	N.A.	R	N.A.	1, 2, 3
f. Trip of Main Feedwater Pumps	N.A.	N.A.	R	1, 2
g. Auxiliary Feedwater Suction Pressure-Low	N.A.	R	M	1, 2, 3
7. LOSS OF POWER				
a. 6.9 kv Shutdown Board Undervoltage				
1. Loss of Voltage	S	R	M	1, 2, 3, 4
2. Load Shedding	S	R	N.A.	1, 2, 3, 4
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS				
a. Pressurizer Pressure, P-11	N.A.	R (4)	N.A.	1, 2, 3
b. T_{avg}, P-12	N.A.	R (4)	N.A.	1, 2, 3
c. Steam Generator Level, P-14	N.A.	R (4)	N.A.	1, 2

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP				
a. RSWT Level - Low COINCIDENT WITH	S	R	M	1, 2, 3, 4
Containment Sump Level - High AND	S	R	M	1, 2, 3, 4
Safety Injection	(See 1 above for all Safety Injection Surveillance Requirements)			

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) The total interlock function shall be demonstrated OPERABLE during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area	1	*	$\leq$ 15 mR/hr	10^{-1} - 10^{4} mR/hr	26
b. Containment Area	1	1, 2, 3 & 4	N/A	1- 10^{7} R/hr***	30
2. PROCESS MONITORS					
a. Containment Purge Air	1	1, 2, 3, 4 & 6	$\leq 8.5 \times 10^{-3}$ μCi/cc	10 - 10^{7} cpm	28
b. Containment					
i. Gaseous Activity					
a)Ventilation Isolation	1	ALL MODES	$\leq 8.5 \times 10^{-3}$ μCi/cc	10 - 10^{7} cpm	28
b)RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{7} cpm	27
ii. Particulate Activity					
a)Ventilation Isolation	1	ALL MODES	$\leq 1.5 \times 10^{-5}$ μCi/cc	10 - 10^{7} cpm	28
b)RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{7} cpm	27
c. Control Room Isolation	1	ALL MODES	$\leq$ 400 cpm**	10 - 10^{7} cpm	29
d. Noble Gas Effluent Monitors					

* With fuel in the storage pool or building

** Equivalent to 1.0×10^{-5} μCi/cc.

*** Measurement range by extrapolation.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 26 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored are with portable monitoring instrumentation at least once per 24 hours.

ACTION 27 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 28 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 29 - With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 30 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel(s) to OPERABLE Status within 7 days, or be in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area	S	R	M	*
b. Containment Area	S	R	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Containment Purge Air Exhaust	S	R	M	1, 2, 3, 4 & 6
b. Containment				
i. Gaseous Activity				
a)Ventilation Isolation	S	R	M	ALL MODES
b)RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity				
a)Ventilation Isolation	S	R	M	ALL MODES
b)RCS Leakage Detection	S	R	M	1, 2, 3 & 4
c. Control Room Isolation	S	R	M	ALL MODES
d. Noble Gas Effluent Monitors				

*With fuel in the storage pool or building.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. 0-XT-52-75A, Containment Elev. 734	0-1.0g	1
b. 0-XT-52-75B, Annulus Elev 680	0-1.0g	1*
c. 0-XR-52-77, Diesel Building Elev. 722	0-1.0g	1
2. Triaxial Peak Accelerographs		
a. 0-XR-52-82, Auxiliary Building Elev. 689	0-5.0g	1
b. 0-XR-52-83, Auxiliary Building Elev. 736	0-5.0g	1
c. 0-XR-52-84, Control Building Elev. 732	0-5.0g	1
3. Biaxial Seismic Switches		
a. 0-XS-52-79, Annulus Elev. 680	0.025-0.25g	1*
b. 0-XS-52-80, Annulus Elev. 680	0.025-0.25g	1*
c. 0-XS-52-81, Annulus Elev. 680	0.025-0.25g	1*
4. Triaxial Response-Spectrum Recorders		
a. 0-XR-52-86, Annulus Elev. 680	2-25.4 Hz, 0.003-90g	1*
b. 0-XR-52-87, Reactor Containment Bldg. Elev. 734	2-25.4 Hz, 0.003-90g	1
c. 0-XR-52-88, Aux. CR Elev. 734	2-25.4 Hz, 0.003-90g	1
d. 0-XR-52-89, DB Bldg. 2A Elev. 713	2-25.4 Hz, 0.003-90g	1

*With reactor control room indication

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelerographs			
a. 0-XT-52-75A, Containment Elev. 734	M*	R	SA
b. 0-XT-52-75B, Annulus Elev. 680**	M*	R	SA
c. 0-XR-52-77, Diesel Building Elev. 722	M*	R	SA
2. Triaxial Peak Accelerographs			
a. 0-XR-52-82, Auxiliary Building Elev. 689	NA	R	NA
b. 0-XR-52-83, Auxiliary Building Elev. 736	NA	R	NA
c. 0-XR-52-84, Control Building Elev. 732	NA	R	NA
3. Biaxial Seismic Switches			
a. 0-XS-52-79, Annulus Elev. 680**	M	R	SA
b. 0-XS-52-80, Annulus Elev. 680**	M	R	SA
c. 0-XS-52-81, Annulus Elev. 680**	M	R	SA
4. Triaxial Response-Spectrum Recorders			
a. 0-XR-52-86**, Annulus Elev. 680	M	R	NA
b. 0-XR-52-87, Reactor Containment Bldg. Elev. 734	NA	R	NA
c. 0-XR-52-88, Aux. CR Elev. 734	NA	R	NA
d. 0-XR-52-89, DB Bldg. 2A Elev. 713	NA	R	NA

*Except seismic trigger
**With reactor control room indications.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant T_{Hot} (Wide Range)	2	1
2. Reactor Coolant T_{Cold} (Wide Range)	2	1
3. Containment Pressure	2	1
4. Refueling Water Storage Tank Level	2	1
5. Reactor Coolant Pressure	2	1
6. Pressurizer Level (Wide Range)	2	1
7. Steam Line Pressure	2/steam line	1/steam line
8. Steam Generator Level - (Wide Range)	1/steam generator	1/steam generator
9. Steam Generator Level - (Narrow Range)	1/steam generator	1/steam generator
10. Auxiliary Feedwater Flow Rate	1/pump	1/pump
11. Reactor Coolant System Subcooling Margin Monitor	1	0
12. Pressurizer PORV Position Indicator*	2/valve	1/valve
13. Pressurizer PORV Block Valve Position Indicator**	2/valve	1/valve
14. Safety Valve Position Indicator	2/valve	1/valve
15. Containment Water Level (Wide Range)	2	1
16. In Core Thermocouples	4/core quadrant	2/core quadrant

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position with power to the valve operator removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant T_{Hot} (Wide Range)	M	R
2. Reactor Coolant T_{Cold} (Wide Range)	M	R
3. Containment Pressure	M	R
4. Refueling Water Storage Tank Level	M	R
5. Reactor Coolant Pressure	M	R
6. Pressurizer Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Level - (Wide)	M	R
9. Steam Generator Level - (Narrow)	M	R
10. Auxiliary Feedwater Flowrate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Pressurizer PORV Position Indicator	M	R
13. Pressurizer PORV Block Valve Position Indicator	M	R
14. Safety Valve Position Indicator	M	R
15. Containment Water Level (Wide Range)	M	R
16. In Core Thermocouples	M	R

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

Fire Zone	Instrument Location	Minimum Instruments Operable Ionization	Photoelectric	Thermal	Infrared
161	SG Blwdn. Rm. El. 734	4			
162	EGTS Rm. El. 734	3			
163	EGTS Rm. El. 734	3			
164	EGTS Fltr. A El. 734		1		
165	EGTS Fltr. A El. 734		1		
166	EGTS Fltr. B El. 734		1		
167	EGTS Fltr. B El. 734		1		
172	Mech. Eqpt. Rm. El. 734	1			
173	Mech. Eqpt. Rm. El. 734	1			
176	480-V Shtdn. Bd. Rm. 1A1 El. 734	2			
188	480-V Shtdn. Bd. Rm. 2A1 El. 734	2			
177	480-V Shtdn. Bd. Rm. 1A1 El. 734	2			
189	480-V Shtdn. Bd. Rm. 2A1 El. 734	2			
178	480-v Shtdn. Bd. Rm. 1A2 El. 734	2			
190	480-V Shtdn. Bd. Rm. 2A2 El. 734	3			
179	480-V Shtdn. Bd. Rm. 1A2 El. 734	2			
191	480-V Shtdn. Bd. Rm. 2A2 EL. 734	3			
180	480-V Shtdn. Bd. Rm. 1B1 El. 734	2			
192	480-V Shtdn. Bd. Rm. 2B1 El. 734	2			
181	480-V Shtdn. Bd. Rm. 1B1 El. 734	2			
193	480-V Shtdn. Bd. Rm. 2B1 El. 734	2			
182	480-V Shtdn. Bd. Rm. 1B2 El. 734	3			
194	480-V Shtdn. Bd. Rm. 2B2 El. 734	2			
183	480-V Shtdn. Bd. Rm. 1B2 El. 734	3			
195	480-V Shtdn. Bd. Rm. 2B2 El. 734	2			
184	6.9-KV Shtdn. Bd. Rm. A El. 734	6			

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

Fire Zone	Instrument Location	Minimum Instruments Operable Ionization	Photoelectric	Thermal	Infrared
69	Mech. Equip. Rm. El. 669			2	
70	Aux. Bldg. A5-A11, Col. W-X, El. 669	5			
71	Aux. Bldg. A5-A11, Col. W-X, El. 669	5			
72	Aux. FW Pump Turbine 1A-S, El. 669	1			
73	Aux. FW Pump Turbine 1A-S, El. 669			1	
76	S.I. & Charging Pump Rms. El. 669			5	
77	S.I. Pump Rm. 1A, El. 669	1			
78	S.I. Pump Rm. 1B, El. 669	1			
79	Charging Pump Rm. 1C, El. 669	1			
80	Charging Pump Rm. 1B, El. 669	1			
81	Charging Pump Rm. 1A, El. 669	1			
88	Aux. Bldg. Corridor A1-A8, El. 669	8			
89	Aux. Bldg. Corridor A1-A8, El. 669	8			
90	Aux. Bldg. Corridor A8-A15, El. 669	8			
91	Aux. Bldg. Corridor A8-A15, El. 669	8			
92	Aux. Bldg. Corridor Col. U-W, El. 669	4			
93	Aux. Bldg. Corridor Col. U-W, El. 669	4			
94	Valve Galley, El. 669	2			
95	Valve Galley, El. 669	2			
39	Cont. Spray Pump 1A-A, El. 653	2			
40	Cont. Spray Pump 1B-B, El. 653	2			
43	RHR Pump 1A-A, El. 653	2			
44	RHR Pump 1B-B, El. 653	2			
47	Aux. Bldg. Corridor, El. 653	10			

TABLE 4.3-8 (Continued)

TABLE NOTATION

* During liquid additions to the tank.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

** During waste gas disposal system operation.

*** During shield building exhaust system operation.

**** During waste gas releases.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. At least two of the reactor coolant loops listed below shall be OPERABLE.

1. Reactor Coolant Loop A and its associated steam generator and Reactor Coolant pump,
2. Reactor Coolant Loop B and its associated steam generator and Reactor Coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and Reactor Coolant pump,
4. Reactor Coolant Loop D and its associated steam generator and Reactor Coolant pump.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required Reactor Coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no Reactor Coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required Reactor Coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required Reactor Coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 21 percent at least once per 12 hours.

4.4.1.2.3 At least one Reactor Coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*All Reactor Coolant pumps may be de-energized for up to 1 hour prov[illegible] (1) no operations are permitted that would cause dilution of the reac[illegible] coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the reactor coolant and/or Residual heat removal (RHR) loops listed below shall be OPERABLE:

1. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,
2. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,
3. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,
4. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,
5. Residual Heat Removal Loop A,
6. Residual Heat Removal Loop B.

b. At least one of the above reactor coolant and/or RHR loops shall be in operation.**

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

**All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10 percent (wide-range indication) at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.

ACTION:

a. With less than the above required RHR/reactor coolant loops OPERABLE, immediately initiate corrective action to return the required RHR/reactor coolant loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The residual heat removal loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation. Four filled reactor coolant loops with at least 2 steam generators having levels greater than or equal to 10 percent (wide-range indication) may be substituted for one RHR loop.

*The normal or emergency power source may be inoperable.

**The RHR pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3.2 All power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) 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 STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. 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 STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel.

4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

4.4.3.2.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1656 cubic feet (equivalent to an indicated level of less than or equal to 92% on the narrow range instrumentation), and at least two groups of presurizer heaters each having a capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY wi h the next 6 hours and in HOT SHUTDOWN within the following 6 hou s.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current at least once per 92 days.

4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The lower containment atmosphere particulate radioactivity monitoring system,

b. The containment pocket sump level monitoring system, and

c. The lower containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. The lower containment atmosphere gaseous and particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment pocket sump level monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 1 GPM leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

SURVEILLANCE REQUIREMENTS (Continued)

a. Monitoring the lower containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment pocket sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing requirements required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
63-560	Accumulator Discharge
63-561	Accumulator Discharge
63-562	Accumulator Discharge
63-563	Accumulator Discharge
63-622	Accumulator Discharge
63-623	Accumulator Discharge
63-624	Accumulator Discharge
63-625	Accumulator Discharge
63-551	Safety Injection (Cold Leg)
63-553	Safety Injection (Cold Leg)
63-557	Safety Injection (Cold Leg)
63-555	Safety Injection (Cold Leg)
63-632	Residual Heat Removal (Cold Leg)
63-633	Residual Heat Removal (Cold Leg)
63-634	Residual Heat Removal (Cold Leg)
63-635	Residual Heat Removal (Cold Leg)
63-641	Residual Heat Removal/Safety Injection (Hot Leg)
63-644	Residual Heat Removal/Safety Injection (Hot Leg)
63-558	Safety Injection (Hot Leg)
63-559	Safety Injection (Hot Leg)
63-543	Safety Injection (Hot Leg)
63-545	Safety Injection (Hot Leg)
63-547	Safety Injection (Hot Leg)
63-549	Safety Injection (Hot Leg)
63-640	Residual Heat Removal (Hot Leg)
63-643	Residual Heat Removal (Hot Leg)
87-558	Upper Head Injection
87-559	Upper Head Injection
87-560	Upper Head Injection
87-561	Upper Head Injection
87-562	Upper Head Injection
87-563	Upper Head Injection
FCV-74-1*	Residual Heat Removal
FCV-74-2*	Residual Heat Removal

*These valves do not have to be leak tested following manual or automatic actuation or flow through the valve.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
T	4°	3.73	1st REFUELING
U	140°	3.73	3
X	220°	3.73	5
Y	320°	3.73	9
S	40°	1.09	EOL
V	176°	1.09	STBY
W	184°	1.09	STBY
Z	356°	1.09	STBY

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A ,ontained borated water volume of between 7857 and 8071 gallons of borated water,

c. Between 1900 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between 385 and 447 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms or by measurement of levels and pressures, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained borated water volume of between 370,000 and 375,000 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 60°F, and

d. A maximum solution temperature of 105°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a, 12 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage less than or equal to 0.01 L_a when the volume between the door seals is pressurized to greater than or equal to 6 psig for at least 15 minutes,

b. By conducting an overall air lock leakage test at not less than P_a (12 psig) and by verifying the overall air lock leakage rate is within its limit:#

   1. At least once per six months, and

   2. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.
*Exemption to Appendix "J" of 10 CFR 50.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours· restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 140 psig when tested pursuant to Specification 4.0.5.

c. At least once per 18 months during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High test signal.

   2. Verifying that each spray pump starts automatically on a Containment Pressure--High-High test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-2 shall be OPERABLE with isolation times as shown in Table 3.6-2.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-2 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-2 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-2 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying that on a Containment Ventilation isolation test signal, each Containment Ventilation Isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-2 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each Containment Purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours, by verifying that when the measure leakage rate of these valves is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

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CONTAINMENT SYSTEMS

CONTAINMENT AIR RETURN FANS

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent containment air return fans shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment air return fan inoperable, restore the inoperable fan to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each containment air return fan shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by:

   1. Verifying that the fan motor current is 28 ± 7.5 amps with the backdraft dampers closed, and

   2. Verifying that with the fan off, the air return fan damper opens when a torque of less than or equal to 68.1 inch-pounds is appied to the counterweight.

b. At least once per 18 months by verifying that the air return fan starts on an auto-start signal after a 10 ± 1 minute delay and operates for at least 15 minutes.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate shutdown boards, and

b. One turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. 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.

SURVEILLANCE REQUIREMENTS

4.7.1.2 In addition to the requirements of Specification 4.0.5 each auxiliary feedwater pump shall be demonstrated OPERABLE by:

a. Verifying that:

1. each motor-driven pump develops a differential pressure of greater than or equal to 1397 psid on recirculation flow.

2. the steam-turbine driven pump develops a differential pressure of greater than or equal to 1183 psid on recirculation flow when the secondary steam supply pressure is greater than 842 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

3. each automatic control valve in the flow path is OPERABLE whenever the auxiliary feedwater system is placed in automatic control or when above 10% of RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal and a low auxiliary feedwater pump suction pressure test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

c. At least once per 7 days by verifying that each non-automatic valve in the auxiliary feedwater system flowpath is in its correct position.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 70°F.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months, during shutdown, by verifying that each component cooling system pump starts automatically on a Safety Injection test signal.

3/4 7.4 ESSENTIAL RAW COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent essential raw cooling water (ERCW) loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one ERCW loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two ERCW loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection test signal.

2. Verifying that each ERCW pump starts automatically on a Safety Injection test signal.

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3/4.7.6 FLOOD PROTECTION PLAN

LIMITING CONDITION FOR OPERATION

3.7.6 The flood protection plan shall be ready for implementation to maintain the plant in a safe condition.

APPLICABILITY: When one or more of the following conditions exist:

a. heavy rainfall conditions in the east Tennessee watershed,

b. an early warning or alert that a critical combination of flood and/or high headwater levels may or have developed,

c. an early warning or alert involving Fontana Dam, or

d. recognizable seismic activity in the east Tennessee region.

ACTION:

a. With a Stage I flood warning issued initiate and complete within 10 hours the Stage I flood protection plan which shall include being in at least HOT STANDBY within 6 hours, with a SHUTDOWN MARGIN of at least 5% delta k/k and T/avg less than or equal to 350°F within the following 4 hours. If within 10 hours following the issuance of a Stage I flood warning communications between the TVA Division of Water Resources and the Sequoyah Nuclear Plant cannot be verified, initiate and complete the Stage II flood protection procedure within the following 17 hours. With a Stage II flood warning issued initiate the Stage II flood protection plan in time to ensure completion before the predicted flooding of the site and no later than 17 hours prior to the predicted arrival time of the initial critical flood level (697 ft msl winter and 703 ft msl summer).

b. With a seismic event occurring after a critical combination of flood and/or headwater alerts are issued verify and maintain communications between TVA Power Control Center and the Sequoyah Nuclear Plant within 6 hours or initiate and complete the Stage I flood protection plan within the following 10 hours. If communications have not been established upon completion of the Stage I flood protection plan initiate and complete the Stage II flood protection plan within the following 17 hours.

ACTION: (Continued)

c. With a Fontana Dam Alert issued verify and maintain communications between Fontana Dam and the Sequoyah Nuclear Plant with 1 hour or initiate and complete the Stage I flood protection plan within 10 hours. If communications have not been established upon completion of the Stage I flood protection plan initiate and complete the Stage II flood protection plan within the following 17 hours.

d. With either the Norris, Cherokee, Douglas, Fort Loudon, Fontana, Hiwassee, Apalachia, Blue Ridge or Tellico dam failed seismically after a critical combination of flood and/or headwater alerts is issued initiate and complete the Stage I flood protection plan within 10 hours. Upon completion of the Stage I flood protection plan initiate and complete the Stage II flood protection plan within the following 17 hours. Both the Stage I and the Stage II flood protection plans will be terminted if it is determined that the potential for flooding the site does not exist.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level in the forebay shall be determined at least once per 8 hours when the water level is less than or equal to 697 feet Mean Sea Level USGS datum during October 1 through April 15, or 703 feet Mean Sea Level USGS datum during April 16 through September 30; and at least once per 2 hours when the water level is above these limits.

4.7.6.2 Communications between Sequoyah Nuclear Plant:

a. and TVA Division of Water Resources shall be maintained every 3 hours during heavy rainfall condition in the east Tennessee watershed.

b. and TVA Power Control Center shall be maintained every 3 hours following a recognizable seismic event that has occurred when a critical combination of flood and/or headwater alert is issued. Communications shall be maintained until it has been determined that the potential for flooding the site does not exist,

c. and Fontana Dam shall be maintained every hour when an alert involving Fontana Dam has been issued by TVA Division of Water Resources.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room emergency ventilation systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4

With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6

a. With one control room emergency ventilation system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode.

b. With both control room emergency air ventilation systems inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.3 are not applicable in MODE 6.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 104°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent auxiliary building gas treatment filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one auxiliary building gas treatment filter train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8 Each auxiliary building gas treatment filter train shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 9000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

   3. Verifying a system flow rate of 9000 cfm + 10% during system operation when tested in accordance with ANSI N510-1975.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All safety-related snubbers shall be OPERABLE. The snubbers are shown in Tables 4.7.9.a and 4.7.9.b and are listed in Surveillance Instruction SNP SI-162. Any exemptions to the surveillance program are shown in Table 4.7.9.c and in SNP SI-162.

APPLICABILITY: Modes 1, 2, 3, and 4. (Modes 5 and 6 for snubbers located on systems or partial systems required OPERABLE in these Modes.)

ACTION: With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5. These snubbers are shown in Tables 4.7.9.a and 4.7.9.b, and are listed in Surveillance Instruction SNP SI-162. Table 4.7.9.b is a detailed tabulation of the hydraulic snubbers which are also shown in Table 4.7.9.a. Any exemption to any portion of the surveillance program for any snubber is shown in Table 4.7.9.c.

a. Inspection Groups

The snubbers may be categorized into two major groups based on whether the snubbers are accessible or inaccessible during reactor operation. These major groups may be further subdivided into subgroups based on design, environment, or other features which may be expected to affect the OPERABILITY of the snubbers within the subgroup. Each subgroup or group may be inspected independently in accordance with 4.7.9.b through 4.7.9.h.

b. Visual Inspection Schedule and Lot Size

The first inservice visual inspection of snubbers shall be completed by October 31, 1981, and shall include all snubbers on safety-related systems. If less than two (2) snubbers are found

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Schedule and Lot Size (Cont'd)

inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

Number of Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period * #
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Performance and Evaluation

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) bolts attaching the snubber to the foundation or supporting structure are secure, and (3) snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm operability.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested, if applicable, in the as-found condition and determined OPERABLE per Specification 4.7.9.e. Hydraulic snubbers with inoperable single or common fluid reservoirs which have uncovered fluid ports shall be declared inoperable. When hydraulic snubbers which have uncovered fluid ports are tested, the tests shall be performed by starting with the piston at the as-found setting and extending the piston rod in the extension mode direction.

* The inspection interval shall not be lengthened more than one step at a time.
# The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

c. Visual Inspection Performance and Evaluation (Cont'd)

Also, snubbers which have been made inoperable as the result of unexpected transients, isolated damage or other such random events, when the provisions of 4.7.9.g and 4.7.9.h have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

d. Functional Test Schedule, Lot Size, and Composition

During each refueling outage, a representative sample of 10% of the total of the safety related snubbers in use in the plant shall be functionally tested either in place or in a bench test.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers within the groups or subgroups. The representative sample should be weighted to include more snubbers from severe service areas such as near heavy equipment. Unless a failure analysis as required by 4.7.9.f indicates otherwise, the sample shall be a composite based on the ratio of each group to the total number of snubbers installed in the plant. Snubbers placed in the same location as snubbers which failed the previous functional test shall be included in the next test lot if the failure analysis shows that failure was due to location.

The security of fasteners for attachment of the snubbers to the component and to the snubber anchorage shall be verified on snubbers selected for functional tests.

e. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers, may be tested to verify only that activation takes place in both directions of travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. Snubber bleed, or release where required, is present in both tension and compression, within the specified range.

3. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel. Also, the increase in the force required shall not exceed 50 percent of the amount required at the last surveillance test of that snubber, provided that the force required is at least 25 pounds.

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

5. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

f. Functional Test Failure Analysis and Additional Test Lots

If any snubber selected for functional testing either fails to lock up or fails to move due to manufacture or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

If more than two snubbers do not meet the functional test acceptance criteria, an additional lot equal to one-half the original lot size shall be functionally tested for each failed snubber in excess of the two allowed failures. An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the operability of other snubbers which may be subject to the same failure mode. (Selection of snubbers for future testing may also be based on the failure analysis.) Testing shall continue until not more than one additional inoperable snubber is found within a subsequent required lot

f. Functional Test Failure Analysis and Additional Test Lots (Cont'd)

or all snubbers of the original inspection group have been tested, or all suspect snubbers identified by the failure analysis have been tested, as applicable.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

Snubbers shall not be subjected to prior maintenance specifically for the purpose of meeting functional test requirements.

g. Functional Test Failure - Attached Component Analysis

For snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are restrained by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components restrained by the snubber(s) were adversely affected by the inoperability of the snubbers(s), and in order to ensure that the restrained component remains capable of meeting the designed service.

h. Functional Testing of Repaired and Spare Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not fail between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information, and the seals shall be replaced so that the maximum expected service life does not expire during a period when the snubber is required to be operable. The seal replacements shall be documented and the documentation shall be retained in accordance with 6.10.2.n.

i. Snubber Service Life Program (Cont'd)

Mechanical snubber drag force increases greater than 50 percent of previously measured values shall be evaluated as an indication of impending failure of the snubber. These evaluations and any associated corrective action shall be documented and the documentation shall be retained in accordance with 6.10.2.n.

j. Exemption From Visual Inspection or Functional Tests

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify snubber operability for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in Table 4.7.9.c and shall continue to be listed in the plant instruction SNP SI-162 indicating the extent of the exemptions.

Table 4.7.9a

Safety Related Snubbers *

	ACCESSIBLE								INACCESSIBLE								
	Small PSA		Medium & Large PSA					Hyd.	Small PSA		Medium & Large PSA					Paul Munroe	Hyd
Size	1/4	1/2	1	3	10	35	100		1/4	1/2	1	3	10	35	100		
MS	22	9		3	9	9	7	12			1					20	16
AMS	1	2						3									
AFD	4	1	1	4				5	1	2		2					2
FD					2			6	1		1		2				1
CC			8	10	8			10	21	5	4	2					4
SI		3			1			2	37	12	2	9	15	1			4
CS				3	4	2		3				1	15	1			
CVC	7	5		1				4	24	7	3	8	1				
RC									15	16	29	40	19	8			
UHI									1	4	7	20	24	5			1
SGB	1		1						1	7	8	5					
FPC			2	4								3	1				
ERCW	2		5		4				22	19	23	15					
RHR	5	2	2	6	2			2			1						3
IC									8	6	5						
WD									9								
DW	1								1		1						
SA									1	1	1						
PW	1	2									1	1					
AC&H									12	7							
Sub Total	44	24	19	31	30	11	7		154	86	87	106	77	15			
Totals		68					98	47		240					285	20	31

*Snubbers may be added to safety related systems without prior License Amendment to Table 4.7.9a provided that a revision to Table 4.7.9a is included with the next License Amendment request. Any exemptions to the provisions of the surveillance program for any snubber is indicated in Table 3.7.9.c.

Table 4.7.9b

Safety Related Hydraulic Snubbers*

ACCESSIBLE

System Size	1	1½	2	2½	3½	4	5	6	8	Sub Total
MS				1	5	5	1			12
AMS		3								3
AFD	2	3								5
FD					3	2		1		6
CC		1	2	3	3	1				10
SI		1				1				2
CS		1				1	1			3
CVC		4								4
RC										0
UHI										0
SGB										0
FPC										0
RHR		2								2
IC										0
WD										0
DW										0
SA										0
PW										0
AC&H										0
ERCW										0
									Total	47

INACCESSIBLE

Size	1	1½	2	2½	3½	4	5	6	8	Sub Total
MS						8			8	6
AMS										0
AFD		2								2
FD					1					1
CC	2	2								4
SI				1		2	1			4
CS										0
CVC										0
RC										0
UHI					1					1
SGB										0
FPC										0
RHR				2	1					3
IC										0
WD										0
DW										0
SA										0
PW										0
AC&H										0
ERCW										0
										0
									Total	31

*Snubbers may be added to safety related systems without prior License Amendment to Table 4.7.9b provided that a revision to Table 4.7.9b is included with the next License Amendment request. Any exemptions to the provisions of the surveillance program for any snubber is indicated in Table 3.7.9.:.

Table 4.7.9c
Safety Related Snubbers - Exemptions to the Surveillance Program

(EXEMPTED SNUBBERS TO BE ADDED LATER.)

Pages 3/4 7-31 through 3/4 7-36a deleted

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and:

1. Either decontaminate and repair the sealed source, or

2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.10.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive materials:

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and plans and schedule for restoring the station to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of all the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 300 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

Table 3.7-5 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
g. CCW Intake Pumping Station		
	690	0-26-866
	690	0-26-867
	690	0-26-868
	690	0-26-869
	690	0-26-870
h. ERCW Pumping Station		
	688	0-26-927
	688	0-26-926
	688	0-26-930
	704	0-26-931
	704	0-26-925
	704	0-26-923
	720	0-26-929
	720	0-26-924
	720	0-26-932

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals with startup on each signal verified at least once per 124 days:

   a) Manual.

   b) Simulated loss of offsite power by itself.

   c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.

   d) An ESF actuation test signal by itself.

5. Verifying the generator is synchronized, loaded to greater than or equal to 4000 kw in less than or equal to 60 seconds, and operates for greater than or equal to 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated shutdown boards.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the engine-mounted fuel tanks.

c. At least once per 92 days and from new fuel oil prior to addition to the 7-day tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @ 100°F of greater than or equal to 1.8 but less than or equal to 5.8 centistokes when tested in accordance with ASTM-D975-77, and an impurity level of less than 2 mg. of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.

d. At least once per 18 months, during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2. Verifying the generator capability to reject a load of greater than or equal to 600 kw while maintaining voltage at 6900 ± 690 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4000 kw without tripping. The generator voltage shall not exceed 7866 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the shutdown boards and load shedding from the shutdown boards.

SURVEILLANCE REQUIREMENTS (Continued)

Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2.c.4. The generator voltage and frequency shall be 6900 ± 690 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 4000 kw.

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its shutdown status.

11. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power.

12. Verifying that the automatic load sequence timers are OPERABLE with the setpoint for each sequence timer within ± 5 percent of its design setpoint.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Engine overspeed

    b) 86 GA lockout relay

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

f. At least once per 10 years* by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypoclorite solution, and

*These requirements are waived for the initial surveillance.

SURVEILLANCE REQUIREMENTS (Continued)

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110 percent of the system design pressure.

4.8.1.1.3 The 125-volt D.C. distribution panel, 125-volt D.C. battery bank and associated charger for each diesel generator shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying:

   1. That the parameters in Table 4.8-1a meet the Category A limits.

   2. That the total battery terminal voltage is greater than or equal to 129-volts on float charge.

b. At least once per 92 days by:

   1. Verifying that the parameters in Table 4.8-1a meet the Category B limits,

   2. Verifying there is no visible corrosion at either terminals or connectors, or the cell to terminal connection resistance of these items is less than 150×10^{-6} ohms, and

   3. Verifying that the average electrolyte temperature of 6 connected cells is above 60 F.

c. At least once per 18 months by verifying that:

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

   2. The battery to battery and terminal connections are clean, tight and coated with anti-corrosion material.

   3. The resistance of each cell to terminal connection is less than or equal to 150×10^{-6} ohms.

4.8.1.1.4 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

TABLE 4.8-1a

DIESEL GENERATOR BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[(1)]	CATEGORY B[(2)]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[(3)] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts[(c)]	> 2.07 volts
Specific Gravity[(a)]	$\geq$ 1.195[(b)]	$\geq$ 1.190	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells $\geq$ 1.190[(b)]

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical boards and inverters shall be OPERABLE and energized with tie breakers open between redundant boards:

6900	Volt Shutdown Board 1A-A
6900	Volt Shutdown Board 1B-B
6900	Volt Shutdown Board 2A-A
6900	Volt Shutdown Board 2B-B
480	Volt Shutdown Board 1A1-A
480	Volt Shutdown Board 1A2-A
480	Volt Shutdown Board 1B1-B
480	Volt Shutdown Board 1B2-B
480	Volt Shutdown Board 2A1-A
480	Volt Shutdown Board 2A2-A
480	Volt Shutdown Board 2B1-B
480	Volt Shutdown Board 2B2-B
120	Volt A.C. Vital Instrument Power Board Channels 1-I and 2-I energized from inverters 1-I and 2-I connected to D.C. Channel I*.
120	Volt A.C. Vital Instrument Power Board Channels 1-II and 2-II energized from inverter 1-II and 2-II connected to D.C. Channel II*.
120	Volt A.C. Vital Instrument Power Board Channels 1-III and 2-III energized from inverter 1-III and 2-III connected to D.C. Channel II
120	Volt A.C. Vital Instrument Power Board Channels 1-IV and 2-IV energized from inverter 1-IV and 2-IV connected to D.C. Channel IV*.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With less than the above complement of A.C. boards OPERABLE and energized, restore the inoperable boards to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one inverter inoperable, energize the associated Vital Instrument Power Board within 8 hours; restore the inoperable inverter to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. boards and inverters shall be determined OPERABLE and energized with tie breakers open between redundant boards at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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*Two invertes may be disconnected from their D.C. source for up to 24 hours for th purpose of performing an equalizing charge on their associated battery bank provic (1) the vital instrument power board is OPERABLE and energized, and (2) the vital instrument power boards associated with the other battery banks are OPERABLE and energized from their respective inverters connected to their respective D.C. sourc

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical boards and inverters shall be OPERABLE and energized:

2 - 6900 volt shutdown boards, either 1A-A and 2A-A or 1B-B and 2B-B,

4 - 480 volt shutdown boards associated with the required OPERABLE 6900 volt shutdown boards,

2 - 120 volt A.C. vital instrument power boards either Channels I and III or Channels II and IV energized from their respective inverters connected to their respective D.C. battery banks, and 480 volt shutdown boards.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. boards and inverters OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. boards and inverters shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated voltage on the bus.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the bus of greater than or equal to 125 volts.

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying that the parameters in Table 4.8-2 meet the Category A limits, and

2. Verifying total battery terminal voltage is greater than or equal to 129-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge (battery terminal voltage below 110-volts), or battery overcharge (battery terminal voltage above 150-volts), by:

1. Verifying that the parameters in Table 4.8-2 meet the Category B limits,

2. Verifying there is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. Verifying that the average electrolyte temperature of 6 connected cells is above 60 F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The resistance of each cell-to-terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 150 amperes at 125-volts for at least 4 hours.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for 2 hours when the battery is subjected to a battery service test.

e. At least once per 60 months by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	$\geq$ 1.200[b]	$\geq$ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195[b]

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

Valve No.	Function	Bypass Device
1-FCV-62-63	Isolation for Seal Water Filter	No
1-FCV-62-138	Safe Shutdown Redundancy (CVCS)	No
1-FCV-62-98	ECCS Operation	No
1-FCV-62-99	ECCS Operation	No
1-FCV-62-90	ECCS Operation	No
1-FCV-62-91	ECCS Operation	No
1-FCV-62-61	Cont. Isolation	No
1-LCV-62-132	ECCS Operation	No
1-LCV-62-133	ECCS Operation	No
1-LCV-62-135	ECCS Operation	No
1-LCV-62-136	ECCS Operation	No
1-FCV-74-1	Open for Normal Plant Cooldown	No
1-FCV-74-2	Open for Normal Plant Cooldown	No
1-FCV-74-3	ECCS Operation	No
1-FCV-74-21	ECCS Operation	No
1-FCV-74-12	RHR Pump, Mini-flow Protects Pump	No
1-FCV-74-24	RHR Pump, Mini-flow Protects Pump	No
1-FCV-74-33	ECCS Operation	No
1-FCV-74-35	ECCS Operation	No
1-FCV-74-7	ECCS Operation	No
1-FCV-74-6	ECCS Operation	No
1-FCV-63-156	ECCS Flow Path	No
1-FCV-63-157	ECCS Flow Path	No
1-FCV-63-39	BIT Injection	No
1-FCV-63-40	BIT Injection	No
1-FCV-63-25	BIT Injection	No
1-FCV-63-26	BIT Injection	No
1-FCV-63-118	RCS Pressure Boundary	No
1-FCV-63-98	RCS Pressure Boundary	No
1-FCV-63-80	RCS Pressure Boundary	No
1-FCV-63-67	RCS Pressure Boundary	No
1-FCV-63-1	ECCS Operation	No
1-FCV-63-72	ECCS Flow Path from Cont. Sump	No
1-FCV-63-73	ECCS Flow Path from Cont. Sump	No
1-FCV-63-8	ECCS Flow Path	No
1-FCV-63-11	ECCS Flow Path	No
1-FCV-63-93	ECCS Cooldown Flow Path	No
1-FCV-63-94	ECCS Cooldown Flow Path	No
1-FCV-63-172	ECCS Flow Path	No
1-FCV-63-5	ECCS Flow Path	No
1-FCV-63-47	Train Isolation	No
1-FCV-63-48	Train Isolation	No
1-FCV-63-4	SI Pump Mini-flow	No
1-FCV-63-175	SI Pump Mini-flow	No

TABLE 3.8-2 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

Valve No.	Function	Bypass Device
1-FCV-63-3	SI Pump Mini-flow	No
1-FCV-63-152	ECCS Recirc	No
1-FCV-63-153	ECCS Recirc	No
1-FCV-63-22	ECCS Recirc	No
1-FCV-3-33	Quick Closing Isolation	No
1-FCV-3-47	Quick Closing Isolation	No
1-FCV-3-87	Quick Closing Isolation	No
1-FCV-3-100	Quick Closing Isolation	No
1-FCV-1-15	Stm Supply to Aux FWP turbine	No
1-FCV-1-16	Stm Supply to Aux FWP turbine	No
1-FCV-3-179A	ERCW Sys Supply to Pump	No
1-FCV-3-179B	ERCW Sys Supply to Pump	No
1-FCV-3-136A	ERCW Sys Supply to Pump	No
1-FCV-3-136B	ERCW Sys Supply to Pump	No
1-FCV-3-116A	ERCW Sys Supply to Pump	No
1-FCV-3-116B	ERCW Sys Supply to Pump	No
1-FCV-3-126A	ERCW Sys Supply to Pump	No
1-FCV-3-126B	ERCW Sys Supply to Pump	No
1-FCV-70-133	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-139	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-4	Isolation for Non-Essential Loads	No
1-FCV-70-143	Isolation for Excess Letdown Ht Xchngr	No
1-FCV-70-92	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-90	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-87	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-89	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-140	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-70-134	Isolation for RCP Oil Coolers & Therm B	No
1-FCV-67-67*	DG Ht Ex	No
2-FCV-67-65*	DG Ht Ex	No
1-FCV-67-66*	DG Ht Ex	No
2-FCV-67-68*	DG Ht Ex	No
1-FCV-67-123	CSS Ht Ex Supply	No
1-FCV-67-125	CSS Ht Ex Supply	No
1-FCV-67-124	CSS Ht Ex Discharge	No
1-FCV-67-126	CSS Ht Ex Discharge	No
0-FCV-67-151*	CCW Ht Ex Throttling	No
0-FCV-67-152*	CCW Ht Ex Throttling	No
2-FCV-67-146	CCW Ht Ex Throttling	No
2-FCV-67-223	Isolation of 1B/2A HDR's	No
1-FCV-67-83	Cont. Isol. Lower	No
1-FCV-67-88	Cont. Isol. Lower	No
1-FCV-67-87	Cont. Isol. Lower	No
1-FCV-67-424*	CCW Ht Ex Isolation	NO
1-FCV-67-478*	Isolation of 1B ERCW HDR	NO

*Common to Units 1&2

TABLE 3.8-2 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

Valve No.	Function	Bypass Device
1-FCV-67-95	Cont. Isol. Lower	No
1-FCV-67-96	Cont. Isol. Lower	No
1-FCV-67-91	Cont. Isol. Lower	No
1-FCV-67-103	Cont. Isol. Lower	No
1-FCV-67-104	Cont. Isol. Lower	No
1-FCV-67-99	Cont. Isol. Lower	No
1-FCV-67-111	Cont. Isol. Lower	No
1-FCV-67-112	Cont. Isol. Lower	No
1-FCV-67-107	Cont. Isol. Lower	No
1-FCV-67-130	Cont. Isol. Upper	No
1-FCV-67-131	Cont. Isol. Upper	No
1-FCV-67-295	Cont. Isol. Upper	No
1-FCV-67-134	Cont. Isol. Upper	No
1-FCV-67-296	Cont. Isol. Upper	No
1-FCV-67-133	Cont. Isol. Upper	No
1-FCV-67-139	Cont. Isol. Upper	No
1-FCV-67-297	Cont. Isol. Upper	No
1-FCV-67-138	Cont. Isol. Upper	No
1-FCV-67-142	Cont. Isol. Upper	No
1-FCV-67-298	Cont. Isol. Upper	No
1-FCV-67-141	Cont. Isol. Upper	No
1-FCV-72-21	Cont. Spray Pump Suction	No
1-FCV-72-22	Cont. Spray Pump Suction	No
1-FCV-72-44	Cont. Spray Pump Suction	No
1-FCV-72-45	Cont. Spray Pump Suction	No
1-FCV-72-2	Cont. Spray Isol.	No
1-FCV-72-39	Cont. Spray Isol.	No
1-FCV-72-40	RHR Cont. Spray Isol.	No
1-FCV-72-41	RHR Cont. Spray Isol.	No

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% delta k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 10 gpm of a solution containing 20,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 One of the following valve combinations shall be verified closed under administrative control at least once per 72 hours:

Combination A	Combination B	Combination C	Combination D
a. 2-81-536	a. 2-81-536	a. 2-81-536	a. 2-81-536
b. 2-62-922	b. 2-62-922	b. 2-62-907	b. 2-62-907
c. 2-62-916	c. 2-62-916	c. 2-62-914	c. 2-62-914
d. 2-62-933	d. 2-62-940	d. 2-62-921	d. 2-62-921
	e. 2-62-696	e. 2-62-933	e. 2-62-940
	f. 2-62-929		f. 2-62-929
	g. 2-62-932		g. 2-62-932
	h. 2-FCV-62-128		h. 2-62-696
			i. 2-FCV-62-128

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

c. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Be capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Containment Ventilation isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the Containment Ventilation isolation valves per the applicable portions of Specification 4.6.3.2

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1. A minimum capacity of 2750 pounds, and

   2. An electrical overload cut off limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1. A minimum capacity of 610 pounds, and

   2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable mainipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 2750 pounds and demonstrating an automatic electrical load cut off when the crane load exceeds 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 610 pounds.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.**

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each RHR loop.
**Prior to initial criticality only one independent RHR loop shall be required OPERABLE.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

TABLE NOTATION

b. Analyses shall also be performed following shutdown from ≥15% RATED THERMAL POWER, startup to ≥15% RATED THERMAL POWER or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 2 days following each shutdown from ≥15% RATED THERMAL POWER, startup to ≥15% RATED THERMAL POWER or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and report.

h. During releases via this exhaust system.

i. In MODES 1, 2, 3 and 4, the upper and lower compartments of the containment shall be sampled prior to VENTING or PURGING. Prior to entering MODE 5, the upper and lower compartments of the containment shall be sampled. The incore instrument room purge sample shall be obtained at the shield building exhaust between 5 and 10 minutes following initiation of the incore instrument room purge..

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in a waste gas holdup tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in a waste gas holdup tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the affected waste gas holdup tank and reduce the concentration of oxygen to less than or equal to 2% by volume within one hour.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by monitoring the waste gas additions to the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

TABLE 4.11-2
RADIOACTIVE GASEOUS WASTE MONITORING SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment Purge	P[i] Each Purge Grab Sample	D[i] Each Purge	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. Noble Gases and Tritium 1. Condenser Vacuum Exhaust[h] 2. Auxiliary Building Exhaust[b,e] 3. Service Building Exhaust 4. Shield Building Exhaust[b,c,h]	M Grab Sample	M	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
D. Iodine and Particulates 1. Auxiliary Building Exhaust 2. Shield Building Exhaust	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
E. Noble Gases all Releases types as listed in A, B, and C above	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitment to any member of the public, due to releases of radioactivity from uranium fuel cycle sources, shall be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem) over 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a 12 consecutive month period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor limit report per Specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 3.5% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The 12 hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, a single reactor coolant loop or residual heat removal (RHR) loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

In MODE 5 single failure considerations require that two RHR loops be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

3/4.4.2 and 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any over-pressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

The power operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide positive shutoff capability should a relief valve become inoperable.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that the plant will be able to control reactor coolant pressure and establish natural circulation conditions.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure 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. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 ESSENTIAL RAW COOLING WATER SYSTEM

The OPERABILITY of the essential raw cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.8 AUXILIARY BUILDING GAS TREATMENT SYSTEM

The OPERABILITY of the auxiliary building gas treatment system ensures that radioactive materials leaking from the ECCS equipment following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing. Cumulative operation of the system with the heaters on for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.7.9 SNUBBERS

Snubbers are designed to prevent unrestrained pipe or component motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping or components as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during relatively low probability events, a period of 72 hours is allowed to replace or restore the inoperable snubber(s) to operable status and perform an engineering evaluation on the supported component or declare the supported system inoperable and follow the appropriate limiting condition for operation statement for that system. The engineering evaluation is performed to determine whether the mode of failure of the snubber has adversely affected any safety-related component or system.

Safety-related snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate fluid level if applicable, and attachment of the snubber to its anchorage. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

3/4.7.9 SNUBBERS (cont'd)

When the cause of the rejection of a snubber in a visual inspection is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible and operability verified by inservice functional testing, if applicable, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. Inspection groups may be established based on design features and installed conditions which may be expected to be generic. Each of these inspection groups are inspected and tested separately unless an engineering analysis indicates the inspection group is improperly constituted. All suspect snubbers are subject to inspection and testing regardless of inspection groupings.

To further increase the assurance of snubber reliability, functional tests shall be performed during each refueling outage. These tests will include stroking of the snubbers to verify proper movement, activation, and bleed or release. The performance of hydraulic snubbers generally depends on a clean, deaerated fluid contained within variable pressure chambers, flowing at closely controlled rates. Since these characteristics are subject to change with exposure to the reactor environment, time, and other factors, their performance within the specified range should be verified. Mechanical snubbers which depend upon overcoming the inertia of a mass and the braking action of a capstan spring contained within the snubber for limiting the acceleration of the attached component (within the load rating of the snubber) are not subject to changes in performance in the same manner as hydraulic snubbers. Pending the development of information regarding the change during the service of the snubber of the acceleration/resistance relationship and the optimum method for detecting this change, these mechanical snubbers may be tested to verify that when subjected to a large change in velocity the resistance to movement increases greatly. The performance change information is to be developed in order to establish test methods to be used during and after the first refueling outage.

Ten percent of the total population of approximately 700 snubbers is an adequate sample for functional tests. The initial sample is to be proportioned among the groups in order to obtain a representative sample. Observed failures of more than two snubbers in the initial lot will require an engineering analysis and testing of additional snubbers selected from snubbers likely to have the same defect. A thorough inspection of the snubber threaded attachments to the pipe or components and the anchorage will be made in conjunction with all required functional tests.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8.1 AND 3/4.8.2 A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies", March 10, 1971, 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137 "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are are based on the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more tha .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES (Continued)

The OPERABILITY of the motor operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.

Circuit breakers actuated by fault currents are used as isolation devices in this plant. The OPERABILITY of these circuit breakers ensures that the IE busses will be protected in the event of faults in non qualified loads powered by the busses.

3/4.11.2.6 GAS DECAY TANKS

Restricting the quantity of radioactivity contained in each gas decay tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure".

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than 0.95 when flooded with unborated water, which includes a conservative allowance of 1.78% delta k/k for uncertainties as described in Section 4.3 of the FSAR.

b. A nominal 10.375 inch center-to-center distance between fuel assemblies placed in the storage racks.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between new fuel assemblies such that k_{eff} will not exceed 0.98 when fuel having a maximum enrichment of 3.5 weight percent U-235 is in place and aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

WITH UNIT 1 IN MODE 5 OR 6 OR DE-FUELED

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	2[b]
STA	1	None

WITH UNIT 1 IN MODES 1, 2, 3 OR 4

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1[a]	None
RO	2[b]	1
AO	2[b]	1
STA	1[a]	None

a/ Individual may fill the same position on Unit 1.

b/ One of the two required individuals may fill the same position on Unit 1.

TABLE 6.2-1 (Continued)

TABLE NOTATION

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 2
SRO - Individual with a Senior Reactor Operators License on Unit 2
RO - Individual with a Reactor Operators License on Unit 2
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 6.2-1 for a period of time not to exceed 2 hours in order to accommodate 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 6.2-1. 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 MODES 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During an absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensing Event Reports and other sources which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five dedicated full-time engineers located onsite.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Assistant Director for Maintenance and Engieering of the Division of Nuclear Power.

6.2.4 SHIFT TECHNICAL ADVISOR (STA)

6.2.4.1 The STA shall serve in an advisory capacity to the shift supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of March 28, 1980 NRC letter to all licensees, except for the Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

*Not responsible for sign-off function.

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for off-control point chemistry conditions,

(vi) Procedures identifying (a) the authority responsible for the interpretation of the data; and (b) the sequence and timing of administrative events required to initiate corrective action, and

(vii) Monitoring of the condensate at the discharge of the condensate pumps for evidence of condenser in-leakage. When condenser in-leakage is confirmed, the leak shall be repaired, plugged, or isolated

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring.

e. Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel,
(ii) Procedures for sampling and analysis,
(iii) Provisions for maintenance of sampling and analysis equipment.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.

2. Cause(s) for the unplanned release.

3. Actions taken to prevent recurrence.

4. Consequences of the unplanned release.

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.14 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Director of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention, Chief of the Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

6.10.1 (continued)

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1 and 6.8.4.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the Operational Quality Assurance Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC, RARC, and the NSRB.

l. Records of analyses required by the radiological environmental monitoring program.

m. Records of secondary water sampling and water quality.

n. Records of the service live monitoring of all hydraulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b including the maintenance performed to renew the service life.

o. Records for Environmental Qualification which are covered under the provisions of Paragraph 2.C.(12)(b) of License No. DPR-77.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

ENCLOSURE 2

JUSTIFICATION

This change in the unit 1 technical specifications (T/S) involves updating the unit 1 T/S to be consistent with the unit 2 T/S. All proposed changes have been previously reviewed and approved by the NRC for unit 2. These changes clarify the T/S and are being submitted at the request of the NRC.