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

UNITED STATES NUCLEAR REGULATORY COMMISSION

LOUISIANA POWER AND LIGHT COMPANY

DOCKET NO. 50-382

NOTICE OF ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U. S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from Paragraph III.D.2(b)(ii) of 10 CFR 50, Appendix J to Louisiana Power and Light Company (the applicant). The applicant has applied for a facility operating license for operation of the Waterford Steam Electric Station, Unit No. 3 (the facility). This facility is a pressurized water reactor located in St. Charles Parish, Louisiana.

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action: The applicant proposes to perform a full pressure air lock test after cold shutdown only when maintenance is performed on the air lock which could affect the air lock sealing capability.

The Need for the Proposed Action: The applicant proposes to substitute the seal leakage test of Paragraph II.D.2(b)(ii) of Appendix J for the full pressure test after cold shutdown. Without the proposed action, either a cumbersome test method must be used or a major design change would be required in order to perform the full pressure air lock test.

Environmental Impacts of the Proposed Action: There are no environmental impacts of the proposed action. Whenever the plant is in cold shutdown (Mode 5) or refueling (Mode 6), containment integrity is not required. However, if an air lock is opened during Modes 5 and 6, paragraph III.D.2(b)(ii) of Appendix J requires that an overall air lock leakage test at not less than P_a be conducted prior to plant heatup and startup (i.e., entering Mode 4). The existing air lock doors are so designed that a full pressure, i.e., P_a (44.0 psig), test of an entire air lock can only be performed after strong backs (structural bracing) have been installed on the inner door. Strong backs are needed since the pressure exerted on the inner door during the test is in a direction opposite to that of the accident pressure direction. Installing strong backs, performing the test, and removing strong backs, requires at least 6 hours per air lock, during which access through the air lock is prohibited. Since the proposed change does not otherwise affect radiological plant effluents nor cause any significant occupational exposures, the Commission concludes that there are no significant radiological environmental impacts associated with this proposed exemption.

If the period 6-month test of paragraph III.D.2(b)(i) of Appendix J and the test required by paragraph III.D.2(b)(iii) of Appendix J are current, no maintenance has been performed on the air lock, and the air lock is properly sealed, there is no reason to expect the air lock to leak excessively, even through it has been opened in Mode 5 or Mode 6.

Accordingly, the staff concludes that the applicant's proposed approach of substituting the seal leakage test of paragraph III.D.2(b)(iii) of the full pressure test of paragrah III.D.2(b)(ii) of Appendix J is acceptable when no maintenance has been performed on an air lock. Whenever maintenance has been performed on the air lock, the requirements of paragrah III.D.2(b)(ii) of Appendix J must still be met by the applicant.

With regard to potential non-radiological impacts, the proposed exemption involves systems located entirely within the restricted area as defined in 10 CFR Part 20. It does not affect non-radiological plant effluents and has no other environmental impact. Therefore, the Commission concludes that there are no significant non-radiological environmental impacts associated with the proposed exemption.

Alternative to the Proposed Action: We have concluded that there is no measurable environmental impact associated with the proposed exemption. The principal alternative would be to deny the requested exemption. This would not reduce environmental impacts of the plant operation.

Alternative Use of Resources: This action does not involve the use of resources not previously considered in connection with the "Final Environmental Statement Related to the operation of Waterford Steam Electric Station, Unit No. 3" dated September 1981.

Agencies and Persons Consulted: The NRC staff reviewed the applicant's request and did not consult other agencies or persons.

FINDINGS OF NO SIGNIFICANT IMPACT

The Commission has determined not to prepare an environmental impact statement for the proposed exemption.

Based upon the foregoing environmental assessment, we conclude that the proposed action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see the Safety Evaluation Report of the staff's review of the applicant's application, as amended, for a Materials License, dated February 9, 1983, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Dated at Bethesda, Maryland this 12th day of December, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Acting Assistant Director
for Licensing
Division of Licensing

NUREG-0973

Technical Specifications

Waterford Steam Electric Station, Unit No. 3

Docket No. 50-382

Appendix "A" to
License No. NPF-26

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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 SHAPE INDEX

1.2 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

AZIMUTHAL POWER TILT - T_q

1.3 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 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

1.5 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

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into 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.

c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY including alarm and/or trip function.

CONTAINMENT INTEGRITY

1.7 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 in compliance with the requirements of 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

1.8 CONTROLLED LEAKAGE shall be the seal water flow supplied from the reactor coolant pump seals.

CORE ALTERATION

1.9 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

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/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

1.11 $\bar{E}$ 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 noniodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES 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

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

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

IDENTIFIED LEAKAGE (Continued)

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.

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, 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

1.18 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 1.2.

PHYSICS TESTS

1.19 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.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.20 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.21 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)

1.22 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low-level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low-level radioactive waste disposal sites.

PURGE - PURGING

1.23 PURGE or PURGING shall be 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.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3390 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 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 electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.27 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,

b. The shield building filtration system is in compliance with the requirements of Specification 3.6.6.1, and

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

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.30 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.31 SOLIDIFICATION shall be the immobilization of wet radioactive wastes such as evaporator bottoms, spent resins, sludges, and reverse osmosis concentrates as a result of a process of thoroughly mixing the waste type with a solidification agent(s) to form a free standing monolith with chemical and physical characteristics specified in the PROCESS CONTROL PROGRAM.

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 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, and

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.36 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be 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. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.38 VENTING shall be 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.

WASTE GAS HOLDUP SYSTEM

1.39 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting 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.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
P	Completed prior to each release.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL MODES

OPERATIONAL MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F> T_{avg}>200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.20.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.20, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation 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 protective instrumentation 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.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Type I Addressable Constants, as listed in Table 2.2-2, are expected to change frequently during plant operation. Type II Addressable Constant values are determined (or confirmed) during PHYSICS TESTS following each fuel loading and are not expected to change during plant operation. Changes to Type I Addressable Constants outside the Allowable Value range require Plant Operations Review Committee review prior to implementation. Changes to Type II Addressable Constants made other than as a result of post-fuel loading PHYSICS TESTS shall require Plant Operations Review Committee review prior to implementation unless the changes are required for Technical Specification Compliance.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant less conservative than the value shown in the Allowable Value column of Table 2.2-2, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
Four Reactor Coolant Pumps Operating	≤ 110.1% of RATED THERMAL POWER	≤ 110.7% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.257% of RATED THERMAL POWER	≤ 0.275% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2365 psia	≤ 2372 psia
5. Pressurizer Pressure - Low	≥ 1684 psia (2)	≥ 1644 psia (2)
6. Containment Pressure - High	≤ 17.1 psia	≤ 17.3 psia
7. Steam Generator Pressure - Low	≥ 764 psia (3)	≥ 748 psia (3)
8. Steam Generator Level - Low	≥ 27.4% (4)	≥ 26.7% (4)
9. Local Power Density - High	≤ 21.0 kW/ft (5)	≤ 21.0 kW/ft (5)
10. DNBR - Low	≥ 1.205 (5)(6)	≥ 1.205 (5)(6)
11. Steam Generator Level - High	≤ 87.7% (4)	≤ 88.4% (4)
12. Reactor Protection System Logic	Not Applicable	Not Applicable
13. Reactor Trip Breakers	Not Applicable	Not Applicable
14. Core Protection Calculators	Not Applicable	Not Applicable
15. CEA Calculators	Not Applicable	Not Applicable
16. Reactor Coolant Flow - Low	≥ 23.8 psid (7)	≥ 23.6 psid (7)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER.

(6) The minimum allowable value of the addressable constant BERR1 in each OPERABLE channel is 1.146.

(7) The setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	$\leq$ 1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2, or 3
63	TR	Azimuthal tilt allowance	$\geq$ 1.02
64	TPC	Thermal power calibration constant	$\geq$ 0.90
65	KCAL	Neutron flux power calibration constant	$\geq$ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The summary statements contained in this section provide the bases for the specifications of Section 2.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

2.1 and 2.2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21.0 kW/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.20 for the CE-1 correlation and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accommodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kW/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1974 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.20 and 21.0 kW/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P "Functional Design Specification for a Core Protection Calculator," January 1981; CEN-148(S)-P "Functional Design Specification for a Control Element Assembly Calculator," January 1981 and Software Change Package - CEN-197(c)-P "CEAC/CPC Software Modifications for Waterford-3 SES," March 1982.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level - High

The Linear Power Level - High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of less than or equal to 110.1% of RATED THERMAL POWER.

Logarithmic Power Level - High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to 0.257% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure - High

The Pressurizer Pressure - High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2365 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure - Low

The Pressurizer Pressure - Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1684 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure - High

The Containment Pressure - High trip provides assurance that a reactor trip is initiated concurrently with a safety injection, a containment isolation, and a main steam isolation. The setpoint for this trip is identical to the ESFAS setpoint.

Steam Generator Pressure - Low

The Steam Generator Pressure - Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level - Low

The Steam Generator Level - Low trip provides protection against events involving a mismatch between steam and feedwater flow. These may be due to a steam or feed line pipe break or other increased steam flow or decreased feed flow events. A large feedwater line break event inside containment establishes the trip setpoint. The setpoint ensures that a trip will occur before the steam generator heat sink is lost. The trip setpoint also ensures that the Reactor Coolant System design pressure will not be exceeded prior to the time emergency feedwater can be supplied for decreased heat removal events such as a loss of condenser vacuum or loss of feedwater flow.

Local Power Density - High

The Local Power Density - High trip is provided to prevent the linear heat rate (kW/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

Local Power Density - High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1845 psia. At this pressure a DNBR - Low trip will automatically occur. This low pressure trip also provides protection against steam generator tube rupture events. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than the fuel design limit such that the decrease

DNBR - Low (Continued)

in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit of 1.20. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	Not more positive than +0.5
d.	Axial Shape Index-Negative	Not more negative than -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1845 psia
f.	Pressurizer Pressure-High	$\leq$ 2355 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	< 4.28
i.	Quality Margin-Low	> 0

Steam Generator Level - High

The Steam Generator Level - High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a steam line break event with a loss-of-offsite power. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a nominal setpoint of 23.8 psid. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/or associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements. Exceptions to these requirements are stated in the individual specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individu-l Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. The combined time interval for any three consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5.15% delta k/k.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.15% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,

2. CEA position,

3. Reactor Coolant System average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading.

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 2.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 2.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron 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 2.0% delta k/k:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

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

   1. Reactor Coolant System boron concentration,
   2. CEA position,
   3. Reactor Coolant System average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration,and
   6. Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.2×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% RATED THERMAL POWER, and

b. Less positive than 0.0×10^{-4} delta k/k/°F whenever THERMAL POWER is > 70% RATED THERMAL POWER, and

c. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching two-thirds of expected core burnup.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{cold}) shall be greater than or equal to 520°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{cold}) less than 520°F, restore T_{cold} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System cold leg temperature (T_{cold}) shall be determined to be greater than or equal to 520°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{cold} is less than 530°F.

#With K_{eff} greater than or equal to 1.0.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 At least one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid makeup tank via either a boric acid makeup pump or a gravity feed connection with one associated heat tracing circuit and charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7a. is OPERABLE, or

b. The flow path from the refueling water storage pool via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage pool in Specification 3.1.2.7b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one 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 is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup tanks is used.

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.

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. One flow path from the boric acid makeup tanks via a boric acid makeup pump with one associated heat tracing circuit and a charging pump to the Reactor Coolant System, or

b. One flow path from the boric acid makeup tanks via a gravity feed valve with one associated heat tracing circuit, and a charging pump to the Reactor Coolant System, or

c. The flow path from the refueling water storage pool via a charging pump 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 2.0% 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 makeup tanks is above the temperature limit line shown on Figure 3.1-1.

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 an SIAS test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. and 3.1.2.2b. delivers at least 40 gpm to the Reactor Coolant System.

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two independent charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 Each required charging pump shall be demonstrated OPERABLE at least once every 18 months by verifying that each charging pump starts in response to an SIAS test signal.

BORIC ACID MAKEUP PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1a. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a., suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a. shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a. is OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a. inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each required boric acid makeup pump shall be demonstrated OPERABLE at least once every 18 months by verifying that each boric acid makeup pump starts in response to an SIAS test signal.

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage pool (RWSP) with:

1. A minimum contained borated water volume of 65,465 gallons (12% indicated level), and

2. A minimum boron concentration of 1720 ppm.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

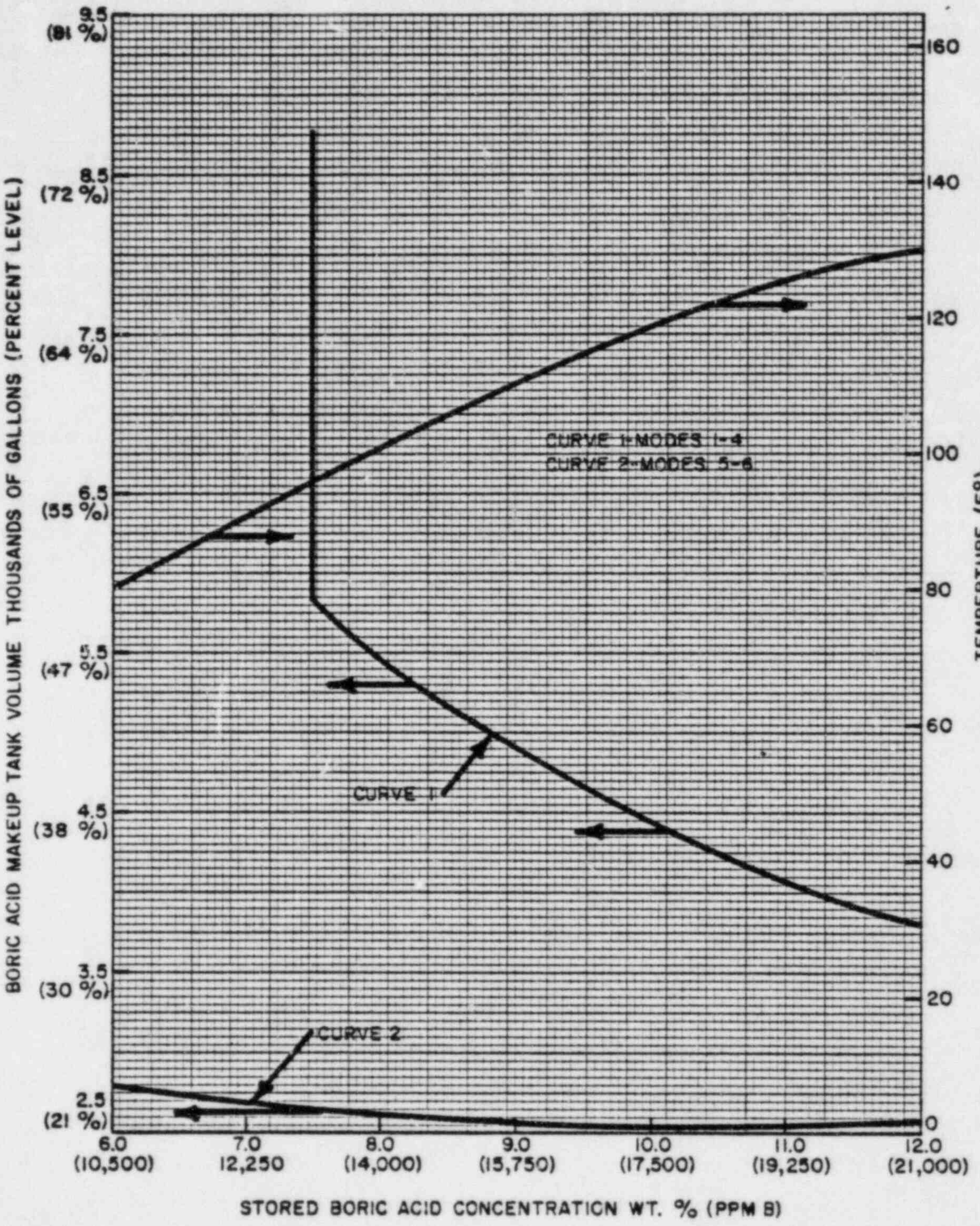

FIGURE 3.1-1

MINIMUM BORIC ACID STORAGE TANK VOLUME AND TEMPERATURE
AS A FUNCTION OF STORED BORIC ACID CONCENTRATION

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank and at least one associated heat tracing circuit per tank with the contents of the tank in accordance with Figure 3.1-1, and

b. The refueling water storage pool with:

1. A minimum contained borated water volume of 475,500 gallons (82% of indicated level), and

2. A boron concentration of between 1720 and 2300 ppm of boron, and

3. A solution temperature between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage pool inoperable, restore the pool to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

2. Verifying the contained borated water volume of the water source, and

3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWSP temperature when the Reactor Auxiliary Building air temperature is less than 55°F or greater than 100°F.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.2.9 Boron concentration shall be verified consistent with SHUTDOWN MARGIN requirements of Specifications 3.1.1.1, 3.1.1.2, and 3.9.1. Boron dilution events shall be precluded by:

a. Either two boron dilution alarms (startup channel high neutron flux) shall be OPERABLE with the alarms set in accordance with Specification 4.1.2.9.5 or the primary makeup water flow path to the Reactor Coolant System shall be isolated, and

b. Removing power to at least two charging pumps in MODE 5 with reactor coolant loops not filled.

APPLICABILITY: MODES 3, 4, 5, and 6.

ACTION:

a. With the boron concentration not consistent with required SHUTDOWN MARGIN, initiate emergency boration.

b. With one boron dilution alarm inoperable and the primary makeup water flow path to the Reactor Coolant System not isolated, determine Reactor Coolant System boron concentration within 1 hour and at least at the monitoring frequency specified in Table 3.1-1.

c. With both boron dilution alarms inoperable and the primary makeup water flow path to the Reactor Coolant System not isolated, determine the Reactor Coolant System boron concentration by two independent means within 1 hour and at least at the monitoring frequency specified in Table 3.1-1; otherwise, immediately suspend all operations involving positive reactivity changes or CORE ALTERATIONS (if applicable).

d. With power applied to more than one charging pump in MODE 5 with the reactor coolant loops not filled, immediately remove power from charging pumps to comply with the above requirement or isolate the primary makeup water flow path to the Reactor Coolant System.

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

SURVEILLANCE REQUIREMENTS

4.1.2.9.1 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 from MODE 2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.2.9.2 Each required boron dilution alarm 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 a CHANNEL CALIBRATION at least once per 18 months.

4.1.2.9.3 The required primary makeup water flow path to the Reactor Coolant System shall be verified to be isolated by either locked closed manual valves, deactivated automatic valves secured in the isolation position, or by power being removed from all charging pumps, at least once per 24 hours.

4.1.2.9.4 At least two charging pumps shall be verified to have power removed when required in MODE 5 with the reactor coolant loops drained at least once per 24 hours.

4.1.2.9.5 Each required boron dilution alarm setpoint shall be adjusted to less than or equal to twice (2x) the existing neutron flux (cps) at the following frequencies:

a. At least once per 5 hours if the reactor has been shut down less than 25 hours;

b. At least once per 24 hours if the reactor has been shut down greater than or equal to 25 hours but less than 7 days;

c. At least once per 7 days if the reactor has been shut down greater than or equal to 7 days.

TABLE 3.1-1

MONITORING FREQUENCIES FOR BORON DILUTION DETECTION

OPERATIONAL MODE	NUMBER OF OPERABLE CHARGING PUMPS* 0	1	2	3
3	24 hr	10 hr	4 hr	3 hr
4	24 hr	8 hr	4 hr	2 hr
5	8 hr	3 hr	1 hr	0.5 hr
5 (System drained for repairs)	8 hr	1 hr	Operation not allowed**	Operation not allowed**
6	24 hr	4 hr	2 hr	1 hr

*Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

**In MODE 5 with the system drained for repairs, at least two charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full-length or part-length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

   Otherwise, be in at least HOT STANDBY within 6 hours.

---

*See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

   Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

f. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements and either greater than or equal to 145 inches withdrawn or within the Long Term Steady State Insertion Limits if in full-length CEA group 6, operation in MODES 1 and 2 may continue.

g. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part-length CEA is maintained within 7 inches (indicated position) of all other part-length CEAs in its group and the CEA is maintained pursuant to the requirements of Specification 3.1.3.7.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core below 145 inches shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other at least once per 12 hours.

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part-length CEA not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*.

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months. The provisions of Specification 4.0.4 are not applicable for performance of this surveillance testing.

*With the reactor trip breakers in the closed position.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.0 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{avg} greater than or equal to 520°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full-length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 145 inches.

APPLICABILITY: MODES 1 and 2*#**.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 145 inches withdrawn, within 1 hour either:

a. Withdraw the CEA to greater than or equal to 145 inches, or

b. Declare the CEA inoperable and determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to greater than or equal to 145 inches withdrawn:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

#With K_{eff} greater than or equal to 1.0.

**Except for surveillance testing pursuant to Specification 4.1.3.1.2.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits* shown on Figure 3.1-2** with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1*** and 2*** #.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, or Reactor Power Cutback, within 2 hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24-hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

---

*Following a reactor power cutback in which (1) Regulating Groups 5 and/or 6 are dropped or (2) Regulating Groups 5 and/or 6 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) are sequentially inserted, the Transient Insertion Limit of Figure 3.1-2 can be exceeded for up to 2 hours.

**CEAs are fully withdrawn in accordance with Figure 3.1-2 when withdrawn to at least 145 inches.

***See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.0.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

   2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

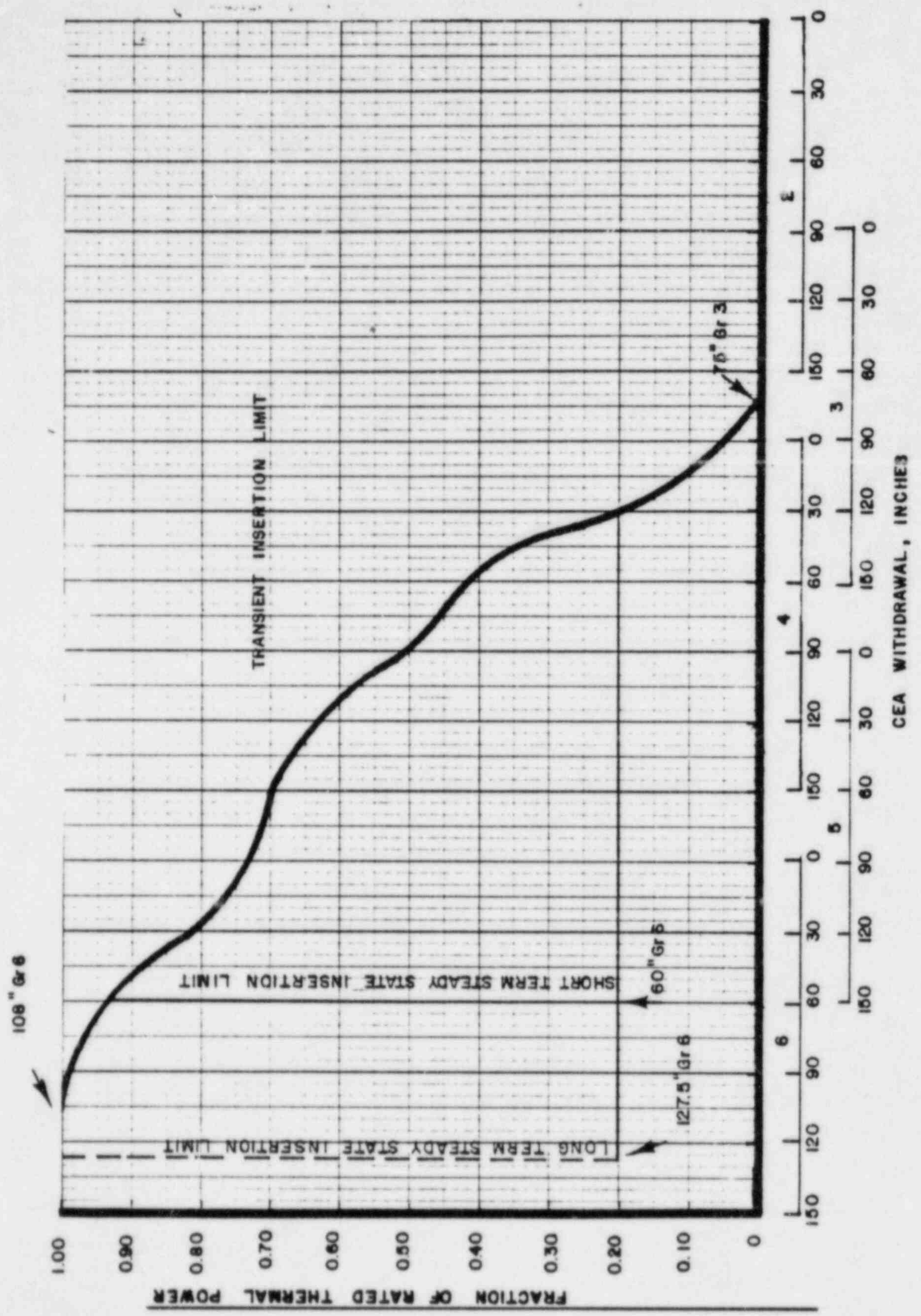

FIGURE 3.1-2
CEA INSERTION LIMITS VS THERMAL POWER

PART-LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 Part-length CEA groups positioned between 0" - 17" withdrawn shall be restricted to prevent the neutron absorber section of the part-length CEA group from covering the same axial segment of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part-length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part-length CEA group to ensure no neutron absorber section of the part-length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part-length CEA group shall be determined at least once per 12 hours.

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kw/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is within the limits shown on Figure 3.2-1.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on kW/ft.

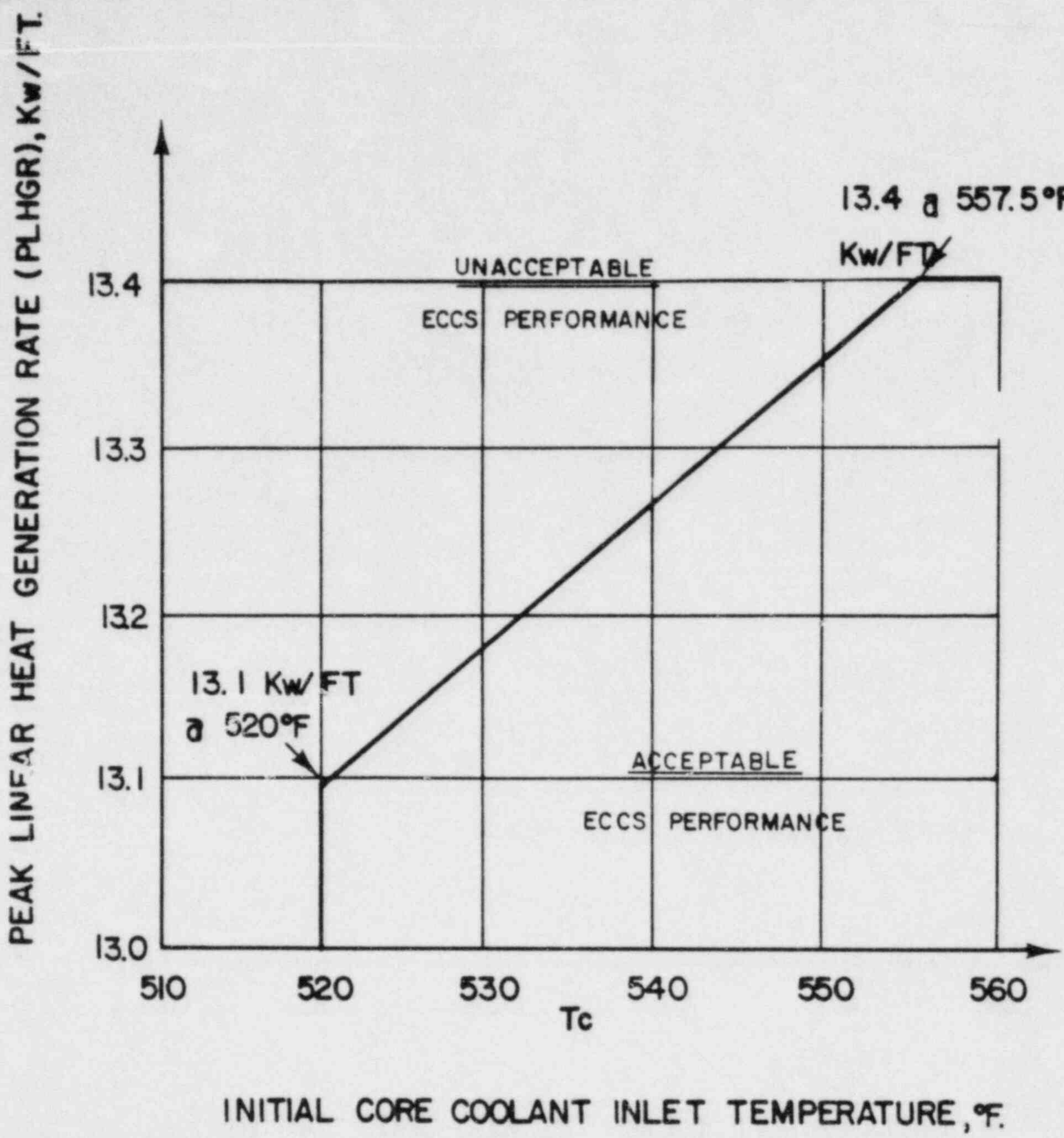

FIGURE 3.2-1

ALLOWABLE PEAK LINEAR HEAT RATE VS Tc

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[F^m_{xy}/F^c_{xy}) - 1.0]$ x 100% is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 effective full power days (EFPD).

*See Special Test Exception 3.10.2.

3/4.2.3 AZIMUTHAL POWER TILT - Tq

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

   1. Due to misalignment of either a part length or full length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1.2 and 4.2.4.2) is detecting the CEA misalignment.

   2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Linear Power Level - High trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-2 or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to increase the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within 1 hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-3.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

BURNUP $(\frac{GWD}{MTU})$	DNBR PENALTY (%)
0-10.0	0.50
10.0-20.0	1.00
20.0-30.0	2.00
30.0-40.0	3.50
40.0-50.0	5.50

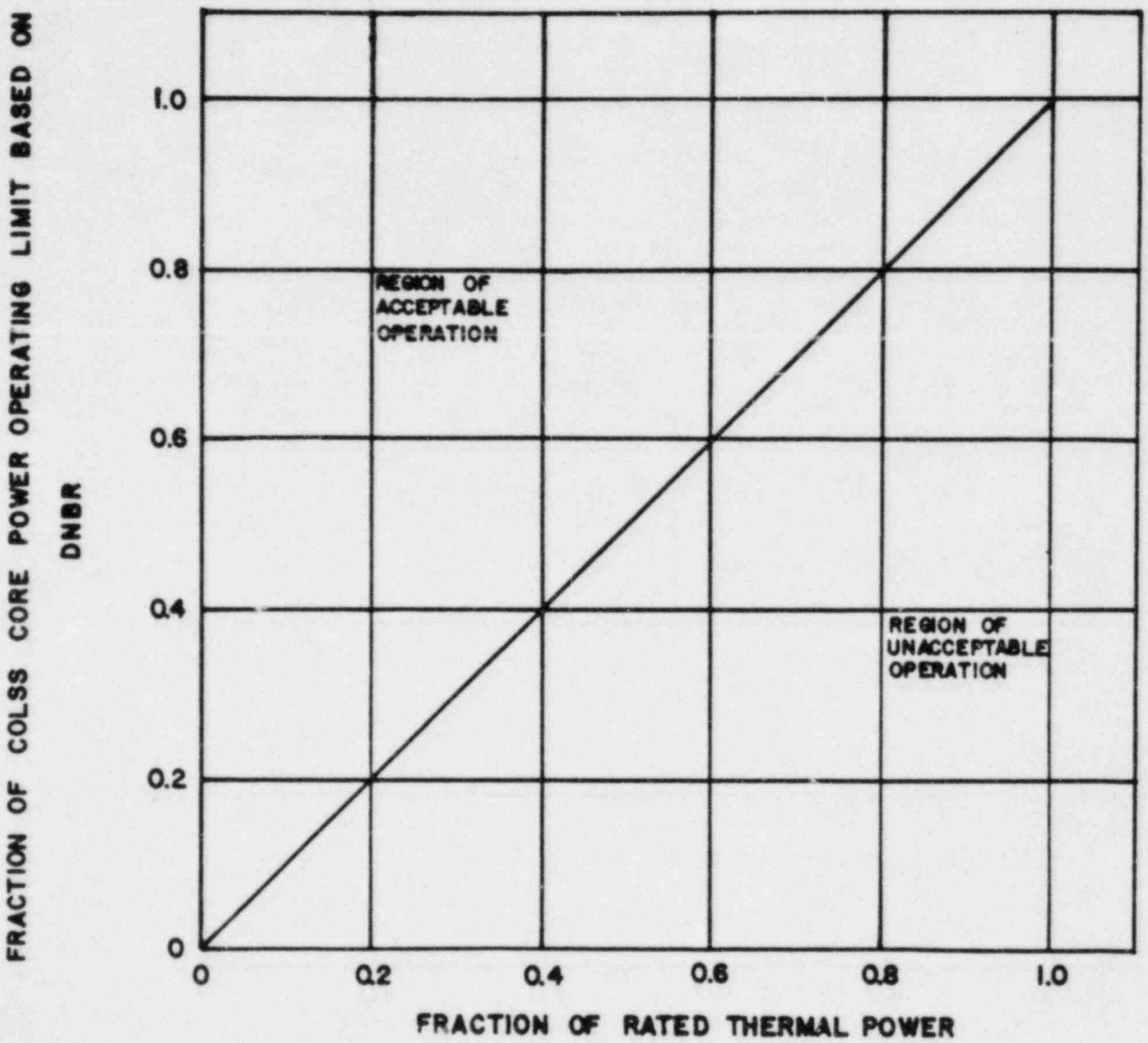

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON COLSS

COLSS OUT OF SERVICE DNBR LIMIT LINE

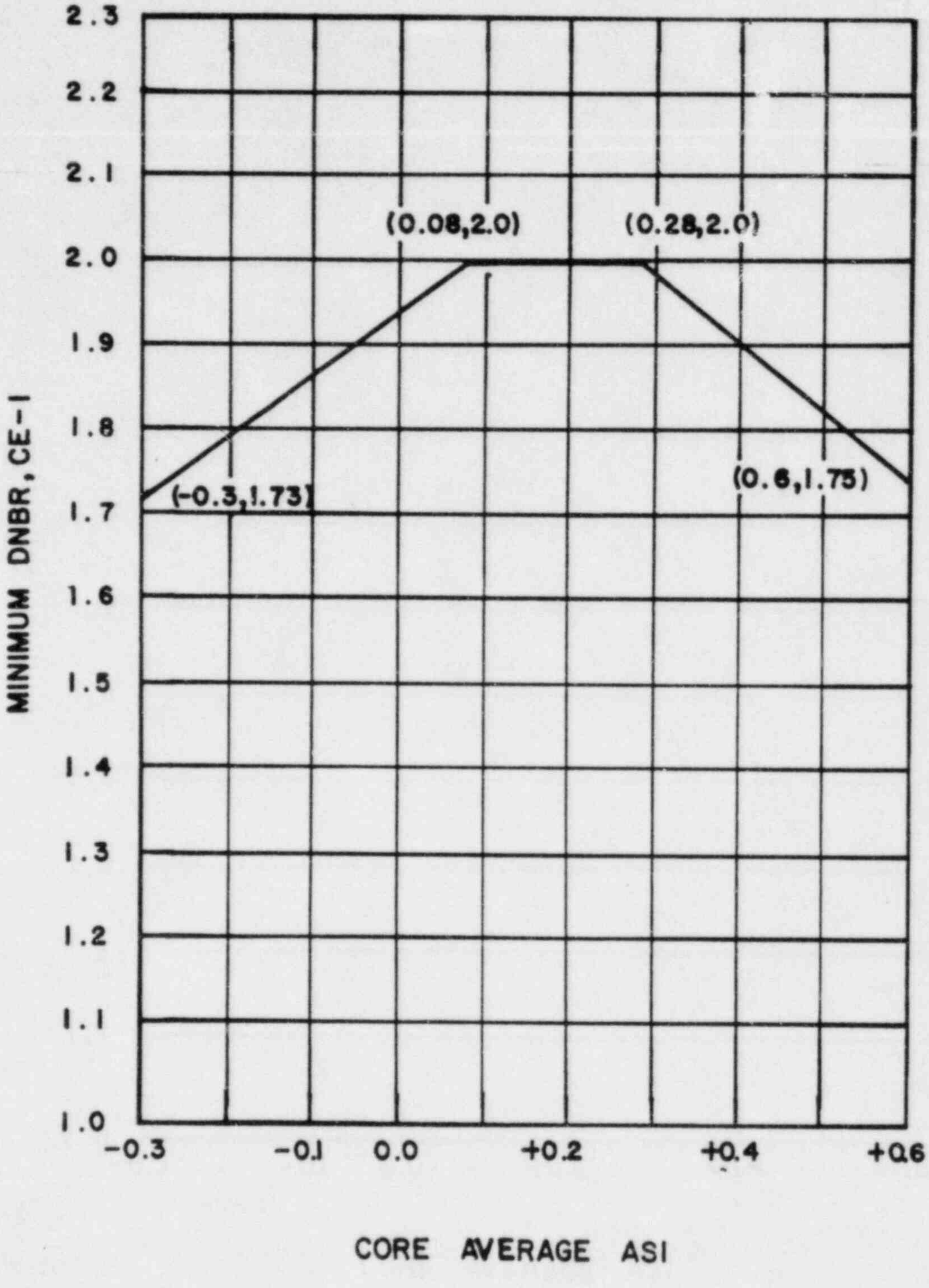

FIGURE 3.2-3

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE)

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 148.0×10^6 lbm/h.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be greater than or equal to the above limit at least once per 12 hours.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be maintained between 544°F and 558°F.*

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*Following a reactor power cutback in which (1) Regulating Groups 5 and/or 6 are dropped or (2) Regulating Groups 5 and/or 6 are dropped and the remaining Regulating Groups (Groups 1, 2, 3, and 4) are sequentially inserted, the upper limit on T_c may increase to 568°F for up to 30 minutes.

POWER DISTRIBUTION LIMITS

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.23 \leq \text{ASI} \leq + 0.50$

b. COLSS OUT OF SERVICE (CPC)
$-0.15 \leq \text{ASI} \leq + 0.50$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With the AXIAL SHAPE INDEX outside its above limits, restore the AXIAL SHAPE INDEX to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The steady-state pressurizer pressure shall be maintained between 2025 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the steady-state pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The steady-state pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses 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 Each reactor protective instrumentation channel 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.

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

4.3.1.3 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 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.

4.3.1.4 The isolation characteristics of each CEA isolation amplifier and each optical isolator for CEA Calculator to Core Protection Calculator data transfer shall be verified at least once per 18 months during the shutdown per the following tests:

a. For the CEA position isolation amplifiers:

1. With 120 volts AC (60 Hz) applied for at least 30 seconds across the output, the reading on the input does not exceed 0.015 volts DC.

SURVEILLANCE REQUIREMENTS (Continued)

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 15.0 volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than 10 megohms when tested using a megohmmeter on the 500 volt DC range.

4.3.1.5 The Core Protection Calculator System and the Control Element Assembly Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.6 The Core Protection Calculator System shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a High CPC Cabinet Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1, 2 3*, 4*, 5*	1 8
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4 4	2(a)(d) 2	3 3	1, 2 3*, 4*, 5*	2#, 3# 8
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2 3*, 4*, 5*	5 8
13. Reactor Trip Breakers	4	2(f)	4	1, 2 3*, 4*, 5*	5 8
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3# and 7
15. CEA Calculators	2	1	2(e)	1, 2	6 and 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

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 at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6k. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low
2. Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3. Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5. Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6. Core Protection Calculator	Local Power Density - High DNBR - Low

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - (RPS) High	Containment Pressure - High Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less those required by the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour; otherwise, 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.

ACTION 6 -
- a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group.
- b. With both CEACs inoperable, operation may continue provided that:
  1. Within 1 hour the margins required by Specification 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 19% of RATED THERMAL POWER and the Reactor Cutback function is disabled, and

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 second*
3. Logarithmic Power Level - High	≤ 0.40 second*
4. Pressurizer Pressure - High	≤ 0.90 second
5. Pressurizer Pressure - Low	≤ 0.90 second
6. Containment Pressure - High	≤ 1.70 seconds
7. Steam Generator Pressure - Low	≤ 0.90 second
8. Steam Generator Level - Low	≤ 0.90 second
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.634 second*
b. CEA Positions	≤ 0.645 second**
c. CEA Positions: CEAC Penalty Factor	≤ 0.429 second
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.634 second*
b. CEA Positions	≤ 0.645 second**
c. Cold Leg Temperature	≤ 0.634 second#
d. Hot Leg Temperature	≤ 0.634 second#
e. Primary Coolant Pump Shaft Speed	≤ 0.487 second**
f. Reactor Coolant Pressure from Pressurizer	≤ 0.634 second##
g. CEA Positions: CEAC Penalty Factor	≤ 0.429 second

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow - Low	0.70 second

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the time the CPC/CEAC receives an input signal until the system outputs a trip signal.

#Response time shall be measured from the output of the sensor. RTD response time for all the RTDs shall be measured at least once per 18 months. The measured P_{τ} of the slowest RTD shall be less than or equal to 6 seconds (P_{τ} assumed in the safety analysis).

##Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.70 second.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R and S/U(1)	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M(10), S/U(1)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),R(4,5)	M(9),R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2
16. Reactor Coolant Flow - Low	S	R	M	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

(1) Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER: adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(10) At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel 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 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least 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 ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	12
b. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
d. Automatic Actuation - Logic	4	2	3	1, 2, 3	12
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	12
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	13*, 14*
c. Automatic Actuation Logic	4	2	3	1, 2, 3	12
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	12
b. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	13*, 14*
d. Automatic Actuation Logic	4	2	3	1, 2, 3	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons)	2 sets of 2 per steam generator	1 set of 2 per steam generator	2 sets of 2 per operating steam generator	1, 2, 3	16
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	13*, 14*
c. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14*
d. Automatic Actuation Logic	4	2	3	1, 2, 3	12
5. SAFETY INJECTION SYSTEM SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Refueling Water Storage Pool - Low	4	2	3	1, 2, 3, 4	13*, 14*
c. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	12
6. LOSS OF POWER (LOV)					
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	3/bus	3/bus	3/bus	1, 2, 3	17
b. 480 V Emergency Bus Undervoltage (Loss of Voltage)	3/bus	3/bus	3/bus	1, 2, 3	17
c. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3/bus	3/bus	3/bus	1, 2, 3	17

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. EMERGENCY FEEDWATER (EFAS)					
a. Manual (Trip Buttons)	2 sets of 2 per steam generator	1 set of 2 per steam generator	2 sets of 2 per steam generator	1, 2, 3	15
b. SG Level (1/2) - Low and ΔP (1/2) - High	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	13*, 14*
c. SG Level (1/2) - Low and No S/G Pressure - Low Trip (1/2)	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	13*, 14*
d. Automatic Actuation Logic	4	2	3	1, 2, 3	12
e. Control Valve Logic (Wide Range SG Level - Low)	2/steam generator	1/steam generator	2/steam generator	1, 2, 3	15

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 400 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 500 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 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 the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6k. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below:

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below.

	Process Measurement Circuit	Functional Unit Bypassed/Tripped
1.	Containment Pressure Circuit	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
3.	Steam Generator Level	Steam Generator Level -Low Steam Generator Level - High Steam Generator ΔP (EFAS)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 13 are satisfied.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - 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.

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the tripped condition within 1 hour, otherwise, comply with the requirements of Action 12.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.3 psia
c. Pressurizer Pressure - Low	$\geq$ 1684 psia(1)	$\geq$ 1644 psia(1)
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	$\leq$ 17.7 psia	$\leq$ 18.0 psia
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.3 psia
c. Pressurizer Pressure - Low	$\geq$ 1684 psia(1)	$\geq$ 1644 psia(1)
d. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 764 psia(2)	$\geq$ 748 psia(2)
c. Containment Pressure - High	$\leq$ 17.1 psia	$\leq$ 17.3 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT			TRIP VALUE	ALLOWABLE VALUES
5.	SAFETY INJECTION SYSTEM SUMP RECIRCULATION (RAS)			
	a.	Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
	b.	Refueling Water Storage Pool - Low	10.0% (57,967 gallons)	9.3% (53,910 gallons)
	c.	Automatic Actuation Logic	Not Applicable	Not Applicable
6.	LOSS OF POWER			
	a.	4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3245 volts	≥ 3245 volts
	b.	480 V Emergency Bus Undervoltage	≥ 372 volts	≥ 354 volts
	c.	4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3640 volts	≥ 3604 volts
7.	EMERGENCY FEEDWATER (EFAS)			
	a.	Manual (Trip Buttons)	Not Applicable	Not Applicable
	b.	Steam Generator (1&2) Level - Low	≥ 27.4%(3) (4)	≥ 26.7%(3) (4)
	c.	Steam Generator ΔP - High (SG-1 > SG-2)	≤ 127.6 psid	≤ 136.6 psid
	d.	Steam Generator ΔP - High (SG-2 > SG-1)	≤ 127.6 psid	≤ 136.6 psid
	e.	Steam Generator (1&2) Pressure - Low	≥ 764 psia(2)	≥ 748 psia(2)
	f.	Automatic Actuation Logic	Not Applicable	Not Applicable
	g.	Control Valve Logic (Wide Range SG Level - Low)	≥ 30.0%(3) (5) ≥ 36.3%(3) (5) (6)	≥ 29.0%(3) (5) ≥ 35.3%(3) (5) (6)

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

(1) Value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 500 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of this distance between steam generator upper and lower level instrument nozzles.

(4) Requires corresponding permissive trip signal of item 7.c., 7.d., or 7.e. to actuate EFAS.

(5) Requires corresponding EFAS trip to actuate control valves.

(6) With SIAS trip signal present.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Shield Building Filtration System	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Safety Injection System Sump Recirculation	Not Applicable
f. EFAS	
Emergency Feedwater Pumps	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	
(1) High Pressure Safety Injection	≤ 30.0*/18.5**
(2) Low Pressure Safety Injection	≤ 45.5*/34.0**
b. Containment Isolation	≤ 23.5*/12.0**
c. Containment Cooling	≤ 31.0*/19.5**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	
(1) High Pressure Safety Injection	≤ 30.0*/18.5**
(2) Low Pressure Safety Injection	≤ 45.5*/34.0**
b. Containment Isolation	≤ 23.5*/12.0**
c. Main Steam Isolation	≤ 4.0*/4.0**
d. Main Feedwater Isolation	≤ 6.0*/6.0**
e. Containment Cooling	≤ 31.0*/19.5**
4. Containment Pressure--High-High	
a. Containment Spray Pump	≤ 15.2*/2.7**
b. Containment Spray Valves	≤ 11.0*/11.0**
c. CCW to RCP Valves	≤ 23.5*/12.0**
5. Containment Area Radiation-High#	
Containment Purge Valves Isolation	≤ 6.2*/6.2**
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ 4.0*/4.0**
b. Main Feedwater Isolation	≤ 6.0*/6.0**
7. Refueling Water Storage Pool-Low	
Containment Sump Recirculation	≤ 120.0*/108.5**
8. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	
Loss of Power (0 volts)	≤ 2***
9. 480V Emergency Bus Undervoltage (Loss of Voltage)	
Loss of Power (0 volts)	N.A.
10. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	
Loss of Power	≤ 11***

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
11. Steam Generator Level-Low	
Emergency Feedwater Pump and Block Valves	$\leq$ 54.0*/42.0**
12. Wide Range Steam Generator Level-Low	
Emergency Feedwater Control Valves	$\leq$ 25.0*/25.0**

NOTE: Response time for all Motor-Driven and Steam-Driven Emergency Feedwater Pumps on all ESF signal starts. $\leq$ 54.0

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

***Response time measured from the sensing relay to the channel output only.

#Response time does not include the detector.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure -- High - High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Pressure - Low	S	R	M	1, 2, 3
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. SAFETY INJECTION SYSTEM RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Refueling Water Storage Pool - Low	S	R	M	1, 2, 3, 4
c. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3, 4
6. LOSS OF POWER (LOV)				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	D(4)	1, 2, 3
b. 480 V Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	D(4)	1, 2, 3
c. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	D(4)	1, 2, 3

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (1/2)-Low and ΔP (1/2) - High	S	R	M	1, 2, 3
c. SG Level (1/2) - Low and No Pressure - Low Trip (1/2)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
e. Control Valve Logic (Wide Range SG Level - Low)	S	R	SA(5)	1, 2, 3

TABLE NOTATION

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

(2) Testing of Automatic Actuation Logic shall include energization/deenergization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays K109, K114, K202, K301, K305, K308 and K313 are exempt from testing during power operation but shall be tested at least once per 18 months and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

(4) Using installed test switches.

(5) To be performed during each COLD SHUTDOWN if not performed in the previous 6 months.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel 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 4.3-3.

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 Fuel Handling Building Ventilation System Isolation	2	*	$\leq$ 100 mR/h	10^{-1} - 10^{4} mR/h	24
b. Containment - Purge & Exhaust Isolation	1/train	1, 2, 3, & 4	$\leq$ 2x background	1 - 10^{5} mR/h	25
2. PROCESS MONITORS					
a. Containment Atmosphere					
1) Gaseous Activity RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	10^{-6} - 10^{-1} μCi/cc	23
2) Particulate Activity RCS Leakage Detection	1	1, 2, 3, & 4	Not Applicable	10^{-11} - 10^{-6} μCi/cc	23
b. Control Room Intake Monitors	1/intake	All MODES	$\leq$ 2x background	10^{-8} - 10^{-2} μCi/cc	26
c. Steam Generator Blowdown Monitor	1	1, 2, 3, & 4	$\leq 10^{-3}$ μCi/cc	10^{-6} - 10^{-1} μCi/cc	28
d. Component Cooling Water System	1/line	All MODES	$\leq 10^{-4}$μCi/cc	10^{-7} - 10^{-2} μCi/cc	28
e. Component Cooling Water System	1	All MODES	$\leq 10^{-4}$ μCi/cc	10^{-7} - 10^{-2} μCi/cc	28

*With irradiated fuel in the storage pool.

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
3. EFFLUENT ACCIDENT MONITORS					
a. Containment High Range	2	1, 2, 3, & 4	Not Applicable	1 - 10^8 R/h	27
b. Plant Stack High Range	1	1, 2, 3, & 4	Not Applicable	10^{-7} - 10^5 μCi/cc	27
c. Condenser Vacuum Pump High Range	1	1, 2, 3, & 4	Not Applicable	10^{-7} - 10^5 μCi/cc	27
d. Fuel Handling Building Exhaust High Range	1	1*, 2*, 3*, & 4*	Not Applicable	10^{-7} - 10^5 μCi/cc	27
e. Main Steam Line High Range	1/steam line	1, 2, 3, & 4	Not Applicable	1 - 10^5 mR/h	27

*With irradiated fuel in the storage pool.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 26 - With the number of channels OPERABLE 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 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2. If the monitor is not restored to OPERABLE status within 7 days after the failure, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, operation of the plant may continue for up to 30 days provided grab samples are taken once per 8 hours and these samples are analyzed for gross activity within 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 Fuel Handling Building Ventilation System Isolation	S	R	M	*
b. Containment - Purge & Exhaust Isolation	S	R	M	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment Atmosphere				
1) Gaseous Activity - RCS Leakage Detection	S	R	M	1, 2, 3, & 4
2) Particulate Activity - RCS Leakage Detection	S	R	M	1, 2, 3, & 4
b. Control Room Intake Monitors	S	R	M	ALL MODES
c. Steam Generator Blowdown Monitor	S	R	M	1, 2, 3, & 4
d. Component Cooling Water System	S	R	M	ALL MODES
e. Component Cooling Water System	S	R	M	ALL MODES

* With irradiated fuel in the storage pool.

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. EFFLUENT ACCIDENT MONITORS				
a. Containment High Range	S	R	M	1, 2, 3, & 4
b. Plant Stack High Range	S	R	M	1, 2, 3, & 4
c. Condenser Vacuum Pump High Range	S	R	M	1, 2, 3, & 4
d. Fuel Handling Building Exhaust High Range	S	R	M	1*, 2*, 3*, & 4*
e. Main Steam Line High Range	S	R	M	1, 2, 3, & 4

*With irradiated fuel in the storage pool.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

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

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments which is accessible during power operation and which is actuated during a seismic event (one or more basemat accelerations of 0.05 g or greater) shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from the accessible actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety. Each of the above seismic monitoring instruments which is actuated during a seismic event (one or more basemat accelerations of 0.05 g or greater) but is not accessible during power operation shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed the next time the plant enters MODE 3 or below. A supplemental report shall then be prepared and submitted to the Commission within 10 days pursuant to Specification 6.9.2 describing the additional data from these instruments.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerograph System		
a. Accelerometer (YT-SM 6000) Adjacent to RB -35 ft MSL	0.02-1.0 g	1
b. Accelerometer (YT-SM 6001) RB +46 ft MSL	0.02-1.0 g	1
c. Accelerometer (YT-SM 6002) Free Field Yard Area	0.02-1.0 g	1
d. Starter Unit (YS-SM 6000) Adjacent to RB -35 ft MSL	0.01-0.02 g	1
e. Starter Unit (YS-SM 6001) RB +51 ft MSL	0.01-0.02 g	1
f. Recorder (YR-SM 6000) Control Room RAB +46 ft MSL	0.02-1.0 g	1
g. Control Unit (YZ-SM 6000) Control Room RAB +46 ft MSL	0.02-1.0 g	1*
h. Playback Unit (YR-SM 6001) Control Room RAB +46 ft MSL	0.02-1.0 g	1
2. Triaxial Peak Accelerographs		
a. YR-SM 6020 RB +3 ft MSL	0-2 g	1
b. YR-SM 6021 RB -4 ft MSL	0-2 g	1
c. YR-SM 6022 RAB +21 ft MSL	0-2 g	1
3. Triaxial Seismic Switches		
a. Seismic Swtich (YS-SM 6060) RB -35 ft MSL	0.1-0.25 g	1
b. Control Unit (YZ-SM 6060) Control Room RAB +46 ft MSL	0.1-0.25 g	1*
4. Triaxial Response-Spectrum Recorders		
a. YR-SM 6040 RB +10 ft MSL	1-32 Hz, 0-2 g	1
b. YR-SM 6041 RAB -35 ft MSL	1-32 Hz, 0-2 g	1
c. YR-SM 6042 RAB +21 ft MSL	1-32 Hz, 0-2 g	1
d. Peak Shock Annunciator (YR-SM 6045) RB -35 ft MSL	1-32 Hz, 0-2 g	1
e. Peak Shock Annunciator Control Unit (YZ-SM 6045) Control Room RAB +46 ft MSL	1-32 Hz, 0-2 g	1

*With reactor control room annunciation.

TABLE 4.3-4
SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelerograph System			
a. Accelerometer (YT-SM 6000) Adjacent to RB -35 ft MSL	N.A.	R	SA
b. Accelerometer (YT-SM 6001) RB +46 ft MSL	N.A.	R	SA
c. Accelerometer (YT-SM 6002) Free Field Yard Area	N.A.	R	SA
d. Starter Unit (YS-SM 6000) Adjacent to RB -35 ft MSL	M	R	SA
e. Starter Unit (YS-SM 6001) RB +51 ft MSL	M	R	SA
f. Recorder (YR-SM 6000) Control Room RAB +46 ft MSL	M	R	SA
g. Control Unit (YZ-SM 6000) Control Room RAB +46 ft MSL	M	R	SA*
h. Playback Unit (YR-SM 6001) Control Room RAB +46 ft MSL	N.A.	R	SA
2. Triaxial Peak Accelerographs			
a. YR-SM 6020 RB +3 ft MSL	N.A.	R	N.A.
b. YR-SM 6021 RB -4 ft MSL	N.A.	R	N.A.
c. YR-SM 6022 RAB +21 ft MSL	N.A.	R	N.A.
3. Triaxial Seismic Switches			
a. Seismic Switch YS-SM 6060 RB -35 ft MSL	M	R	SA
b. Control Unit YZ-SM 6060 Control Room RAB +46 ft MSL	M	R	SA*
4. Triaxial Response-Spectrum Recorders			
a. YR-SM 6040 RB +10 ft MSL	N.A.	R	N.A.
b. YR-SM 6041 RAB -35 ft MSL	N.A.	R	N.A.
c. YR-SM 6042 RAB +21 ft MSL	N.A.	R	N.A.
d. Peak Shock Annunciator YR-SM 6045 RB -35 ft MSL	N.A.	R	N.A.
e. Peak Shock Annunciator Control Unit YZ-SM 6045 Control Room RAB +46 ft MSL	N.A.	R	SA

*With reactor control room annunciation.

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT*		LOCATION (Nominal Elevation)	MINIMUM INSTRUMENTS OPERABLE
1. WIND SPEED			
	a. Primary	33 ft (10 m)	1-(a or b)
	b. Secondary	33 ft (10 m)	
	c. Primary	199 ft (60 m)	1
2. WIND DIRECTION (SIGMA THETA)**			
	a. Primary	33 ft (10 m)	1-(a or b)
	b. Secondary	33 ft (10 m)	
	c. Primary	199 ft (60 m)	1
3. TEMPERATURE DIFFERENCE			
	a. Primary	33 ft - 199 ft (10 m-60 m)	1-(a, b, or c)
	b. Secondary	33 ft - 199 ft (10 m-60 m)	
	c. Primary	33 ft - 199 ft (10 m-60 m)	

*Primary, Secondary - Refers to the tower on which instrument is located, see Specification 5.5.

**Derived from instantaneous wind direction measurements.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT*	LOCATION (Nominal Elevation)	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED			
a. Primary	33 ft (10 m)	D	SA
b. Secondary	33 ft (10 m)	D	SA
c. Primary	199 ft (60 m)	D	SA
2. WIND DIRECTION (SIGMA THETA)**			
a. Primary	33 ft (10 m)	D	SA
b. Secondary	33 ft (10 m)	D	SA
c. Primary	199 ft (60 m)	D	SA
3. TEMPERATURE DIFFERENCE			
a. Primary	33 ft - 199 ft (10 m-60 m)	D	SA
b. Secondary	33 ft - 199 ft (10 m-60 m)	D	SA
c. Primary	33 ft - 199 ft (10 m-60 m)	D	SA

*Primary, Secondary - Refers to the tower on which instrument is located, see Specification 5.5.

**Derived from instantaneous wind direction measurements.

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE, with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown system instrumentation channels less than required by Table 3.3-9, either restore the inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5. Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

TABLE 3.3-9

REMOTE SHUTDOWN INSTRUMENTATION

INSTRUMENTATION	READOUT LOCATION	MINIMUM CHANNELS OPERABLE
1. Neutron Flux	LCP-43	2
2. Reactor Trip Breaker Indication	Switch Gear Area	1/trip breaker
3. Reactor Coolant Temperature - Cold Leg (T_{Cold})	LCP-43	2
4. Reactor Coolant Temperature - Hot Leg (T_{Hot})	LCP-43	2
5. Pressurizer Pressure	LCP-43	2
6. Pressurizer Level	LCP-43	2
7. Steam Generator Level	LCP-43	2/steam generator
8. Steam Generator Pressure	LCP-43	2/steam generator
9. Shutdown Cooling Flow Rate	LCP-43	1/train
10. Emergency Feedwater Flow Rate	LCP-43	1/steam generator
11. Condensate Storage Pool Level	LCP-43	2

TABLE 4.3-6

REMOTE SHUTDOWN INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Neutron Flux	M	R*
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Temperature-Cold Leg (T_{Cold})	M	R
4. Reactor Coolant Temperature - Hot Leg (T_{Hot})	M	R
5. Pressurizer Pressure	M	R
6. Pressurizer Level	M	R
7. Steam Generator Level	M	R
8. Steam Generator Pressure	M	R
9. Shutdown Cooling Flow Rate	M	R
10. Emergency Feedwater Flow Rate	M	R
11. Condensate Storage Pool Level	M	R

*Neutron detector may be excluded from CHANNEL CALIBRATION.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 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 channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 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 NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Generator Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator**	1/steam generator**
9. Refueling Water Storage Pool Water Level	2	1
10. Emergency Feedwater Flow Rate	1/steam generator**	1/steam generator**
11. Reactor Cooling System Saturation Margin Monitor	2	1
12. Safety Valve Position Indicator	1/valve	1/valve
13. Containment Water Level (Narrow Range)	1***	1***
14. Containment Water Level (Wide Range)	2	1
15. Core Exit Thermocouples	4/core quadrant	2/core quadrant
16. Containment Isolation Valve Position Indicators*	1/valve#	1/valve#
17. Condensate Storage Pool Level	2	1

#If the containment isolation valve is declared inoperable and the provisions of Specification 3.6.3 are complied with, position indicators may be inoperable; otherwise, comply with the provisions of Specification 3.3.3.6.

*Containment isolation valves listed in Table 3.6-2 (Category 1).

**These corresponding instruments may be substituted for each other.

***Operation may continue for up to 30 days with less than the Minimum Channels OPERABLE requirement.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature -T_{Cold} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Pool Water Level	M	R
10. Emergency Feedwater Flow Rate	M	R
11. Reactor Coolant System Saturation Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Containment Water Level (Narrow Range)	M	R
14. Containment Water Level (Wide Range)	M	R
15. Core Exit Thermocouples	M	R
16. Containment Isolation Valve Position	M	R
17. Condensate Storage Pool Level	M	R

INSTRUMENTATION

CHEMICAL DETECTION SYSTEMS

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.7.1 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 3 ppm, shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

a. With one chlorine detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

CHEMICAL DETECTION SYSTEMS

AMMONIA DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.7.2 Two independent ammonia detection systems, with their alarm/trip setpoints adjusted to actuate at an ammonia concentration of less than or equal to 50 ppm, shall be OPERABLE.

APPLICABILIITY: All MODES.

ACTION:

a. With one ammonia detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

b. With no ammonia detection system OPERABLE, within 1 hour initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.2 Each ammonia detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment or annulus, then inspect that containment or annulus zone at least once per 8 hours or for the containment, monitor air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment or annulus, then inspect that containment or annulus zone at least once per 8 hours or for the containment, monitor air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months. Circuits which are not accessible during plant operation shall be demonstrated OPERABLE during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

SURVEILLANCE REQUIREMENTS (Continued)

4.3.3.8.3 The nonsupervised circuits associated with detector alarms between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.8.4 Each of the resistor wires required by Table 3.3-11 shall be demonstrated OPERABLE at least once per 6 months by verifying the proper wire resistance.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

ZONE	ROOM NAME/NUMBER	ELEVATION (ft)	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1. REACTOR AUXILIARY BUILDING					
RAB 1A	Control Room Proper/304	+46			20/0
RAB 1B	Emergency Equip. H&V Room/314	+46			0/12
RAB 1D	Computer Room (above raised floor)/306	+46			5/0
	Computer Room (below raised floor)/306	+46			0/7
RAB 2	Ventilation Equip. Room/299	+46			0/36
RAB 3	RAB Corridor to Relay Room/261	+35	0/1(3)		4/0
	RAB HVAC Switchgear Equip. Room/323	+46			0/10
RAB 3A	RAB Battery Exhaust Fan Room/406	+69			0/2
RAB 4	Cable Vault/260	+35			0/27
RAB 5	Electrical Penetration Area "A"/263	+35			0/13
RAB 6	Electrical Penetration Area "B"/263A	+35			0/14
RAB 7	Relay Room/262	+35	(3)		12/0
	Isolation Panels (9 Compartments - 2 per comp.)	+35			2/0
RAB 8A	High Voltage Switchgear Room "A"/212A	+21	0/1(1)		18/0
RAB 8B	Electrical Equip. Room/225B and High Voltage Switchgear Room "B"/212	+21	0/1(2)		28/0
	480V Switchgear 3A32 Room	+21	(2)		2/0
RAB 8C	High Voltage Switchgear Room "A-B"/212B	+21	(1)		8/0
RAB 8E	CEA M/G Set Room/216	+21			2/0
RAB 9	Remote Shutdown Panel Room/217	+21			1/0
RAB 11	Battery Room "B"/213	+21			2/0
RAB 12	Battery Room "AB"/214A	+21			2/0
RAB 13	Battery Room "A"/214	+21			2/0
RAB 15	Emergency Diesel Gen. "B" Room/222	+21	0/1		
RAB 15A	Emergency Diesel Gen. "B" Feed TK Room/328A	+46	0/1		
RAB 16	Emergency Diesel Gen. "A" Room/221	+21	0/1		
RAB 16A	Emergency Diesel Gen. "A" Feed Tk. Room 328A	+46	0/1		
RAB 17	CCW Heat Exchanger "B"/236	+21			0/4
RAB 18	CCW Heat Exchanger "A"/220	+21			0/4
RAB 19	CCW Pump "A"/235	+21			2/0
RAB 20	CCW Pump "AB"/234	+21			0/2
RAB 21	CCW Pump "B"/233	+21			1/0
RAB 23	Corridor to CCW Pumps/218, Corridor to CCW Heat Exchangers/219 and Corridor to Emergency Diesel Gen./225A	+21			0/39

(1)Common Resistor Wire
(2)Common Resistor Wire
(3)Common Resistor Wire

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

ZONE	ROOM NAME/NUMBER	ELEVATION (ft)	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1. REACTOR AUXILIARY BUILDING (Continued)					
RAB 25	Equip. Access Area/226 (wing area)	+21			15/0
RAB 27A	H&V Room/124	+ 7			0/6
RAB 27B	Electrical Area and Health Physics Offices/122	+ 7			0/35
RAB 27C	I&C Room/120	+ 7			0/2
RAB 27D	Communications Equip. Room/123	+ 7			1/0
RAB 31	Pipe Chase	- 4			1/0
	Corridor's and Passageways	- 4			0/24
	Corridors on eastside	- 4			0/21
RAB 32	Wing Area westside - Auxiliary Component Cooling Water Pump "A"/B53 and Pipe Penetration Area/B100	-35 & - 4			32/0
	Wing Area Center/B53 and B100	-35 & - 4			28/0
	Wing Area eastside-Component Cooling Water Pump "B"/B53 and Pipe Penetration Area/B100	-35 - 4			31/0
RAB 33	S/D Cooling Heat Exchangers A&B/B20 & B48	-35			0/18
RAB 34	Valve Operating Enclosure Bay Room "A"/B54	-15.5			2/0
	Valve Operating Enclosure Bay Room "B" B55A	-15.5			4/0
RAB 35	Safety Injection Pump Room B/B16	-35			10/0
RAB 36	Safety Injection Pump Room A/B15	-35			10/0
RAB 37	Motor-Driven Emergency Feedpump "A"/B49A	-35			0/1
RAB 38	Motor-Driven Emergency Feedpump "B"/B49B	-35			1/0
RAB 39	General Equipment Area/B5, 12, 13, & 49	-35			0/10
	Corridors & General Equip. Areas/B5, 1, 2, 3, 4, 39, 40, 41, 42, 44 & 46	-35			0/28
	East Corridor & General Equip. Areas/ B17, 23 & 25	-35			0/15
	BA Make-up Tank "A"/B38	-35			4/0
	BA Make-up Tank "B"/B53A	-35			4/0
RAB 40	Diesel Storage Tank "A"/B50	-35			3/0
RAB 41	Diesel Storage Tank "B"/B52	-35			3/0

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

ZONE	ROOM NAME/NUMBER	ELEVATION (ft)	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
2. REACTOR CONTAINMENT BUILDING**					
RCB 1-1	Annulus/420	- 4, +21 & +46			69/0
RCB 2	Electrical Penetration Area A	+21			24/0
RCB 3	Electrical Penetration Area B	+21			21/0
RCB 4	Reactor Cable Trays	+46			16/0
CT 1&3	Wet & Dry Cooling Tower "A" Cable Tray		1/0		
CT 2&4	Wet & Dry Cooling Tower "B" Cable Tray		1/0		
3. FUEL HANDLING BUILDING					
FHB 2	Purification Pump Room/B155, Fuel Pool Pump "A"/B157, Fuel Pool Pump "B"/ B156, Fuel Pool Heater Exchanger/ B158 and Access Area/B-161	+ 1			10/0
	Emergency Filter Train Unit/B152	+ 1			6/0
	Emergency Elect. Equip. Room/B151	+ 1			1/0
FHB 4	Operating Floor/361	+46			15/0
4. CHARCOAL AIR FILTER UNITS					
E-35 (3A-5A)	FHB Emergency Filter Train Unit	+ 1	1/0		
E-35 (3B-5B)	FHB Emergency Filter Train Unit	+ 1	1/0		
E-17 (3A-5A)	Shield Building Ventilation System "A"	+46	1/0		
E-17 (3B-5B)	Shield Building Ventilation System "B"	+46	1/0		
E-23 (3A-5A)	Controlled Ventilation Area System Filter Train "A"	+46	1/0		
E-23 (3A-5B)	Controlled Ventilation Area System Filter Train "B"	+46	1/0		
S-8 (3A-5A)	Control Room Emergency Unit "A"	+46	1/0		
S-8 (3B-5B)	Control Room Emergency Unit "B"	+46	1/0		

TABLE NOTATIONS

*(x/y): x is the number of Function A (early warning fire detection and notification only)instruments.
y is the number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

**The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

INSTRUMENTATION

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

a. a CHANNEL CHECK at least once per 24 hours,

b. a CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. a CHANNEL CALIBRATION at least once per 18 months.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, explain in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8, why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Boric Acid Condensate Discharge	1	28
b. Waste, Waste Condensate and Laundry Discharge	1	28
c. Dry Cooling Tower Sumps	1/sump	29
d. Turbine Building Industrial Waste Sump	1	29
2. RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Circulating Water Discharge (Blowdown Heat Exchanger and Auxiliary Component Cooling Water Pumps)	1	29
3. FLOW RATE MEASUREMENT DEVICES		
a. Boric Acid Condensate Discharge	1	30
b. Waste, Waste Condensate and Laundry Discharge	1	30
c. Turbine Building Industrial Waste Sump*	N.A.	N.A.
d. Dry Cooling Tower Sumps*	N.A.	N.A.
e. Circulating Water Discharge* (Blowdown Heat Exchanger and Auxiliary Component Cooling Water Pumps)	N.A.	N.A.

TABLE 3.3-12 (Continued)

TABLE NOTATIONS

*Pump performance curves generated in place shall be used to estimate flow.

ACTION STATEMENTS

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for radioactivity at a lower limit of detection of at least 10-7 microcurie/mL.

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram DOSE EQUIVALENT I-131.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. RADIOACTIVITY MONITORS PROVIDING ALARMS AND AUTOMATIC TERMINATION OF RELEASE				
a. Boric Acid Condensate Discharge	P	P	R(3)	Q(1)
b. Waste, Waste Condensate and Laundry Discharge	P	P	R(3)	Q(1)
c. Dry Cooling Tower Sumps	D	M	R(3)	Q(5)
d. Turbine Building Industrial Waste Sump	D	M	R(3)	Q(5)
2. RADIOACTIVITY MONITORS PROVIDNG ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Circulating Water Discharge (Blowdown Heat Exchanger and Auxiliary Component Cooling Water Pumps)	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Boric Acid Condensate Discharge	D(4)	N.A.	R	Q
b. Waste, Waste Condensate and Laundry Discharge	D(4)	N.A.	R	Q
c. Turbine Building Industrial Waste Sump	N.A.	N.A.	N.A.	N.A.
d. Dry Cooling Tower Sumps	N.A.	N.A.	N.A.	N.A.
e. Circulating Water Discharge (Blowdown Heat Exchangers and Auxiliary Component Cooling Water Pumps)	N.A.	N.A.	N.A.	N.A.

TABLE 4.3-8 (Continued)

TABLE NOTATION

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

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

2. Circuit failure.

3. Instrument indicates a downscale 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 for 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 contiuous, periodic, or batch releases are made.

(5) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the alarm/trip setpoint and that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm set.

2. Circuit failure.

3. Instrument controls not set in operate mode.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, explain in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8, why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1.	WASTE GAS HOLDUP SYSTEM			
	a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	35
	b. Effluent System Flow Rate Measuring Device	1	*	36
2.	WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
	a. Hydrogen Monitor	1	**	38
	b. Oxygen Monitors	2	**	40
3.	MAIN CONDENSER EVACUATION AND TURBINE GLAND SEALING SYSTEM			
	a. Noble Gas Activity Monitor	1	*	37
	b. Iodine Sampler#	1	*	39
	c. Particulate Sampler#	1	*	39
	d. Sampler Flow Rate Monitor	1	*	36

#If a primary to secondary leak exists or if the noble gas monitors in the main condenser evacuation and turbine gland sealing system or if the steam generator blowdown monitor indicates the presence of radioactivity in the secondary system, the flow from this release point shall be diverted immediately to the plant stack. These instruments are in the plant stack and sampling for radioiodines and particulates shall occur at the plant vent when this occurs.

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. REACTOR AUXILIARY BUILDING VENTILATION SYSTEM (PLANT STACK)			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release#	1	*	37
b. Iodine Sampler	1	*	39
c. Particulate Sampler	1	*	39
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Monitor	1	*	36
5. FUEL HANDLING BUILDING VENTILATION SYSTEM (NORMAL)			
a. Noble Gas Activity Monitor	1	***	37
b. Iodine Sampler	1	***	39
c. Particulate Sampler	1	***	39
d. Flow Rate Monitor	1	***	36
e. Sampler Flow Rate Monitor	1	***	36

#Automatic termination of containment purge only.

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

*At all times.

**During WASTE GAS HOLDUP SYSTEM operation.

***With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

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

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours. For the waste gas holdup tank this action item is applicable only during periods of release. For the main condenser evacuation and turbine gland sealing systems, this action item applies only during release via the discharge silencer and only during turbine gland sealing operations and/or vacuum pump operation.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours. However, containment purging of radioactive effluents must be immediately suspended during this condition for the plant stack only.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the WASTE GAS HOLDUP SYSTEM may continue for up to 14 days provided grab samples are collected at least once per 8 hours and analyzed within the following 4 hours for the onservice gas decay tank.

TABLE 3.3-13 (Continued)

ACTION STATEMENTS

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 40 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of the WASTE GAS HOLDUP SYSTEM may continue provided that the system is sampled by either the remaining monitor or by a grab sample once per 4 hours and the oxygen concentration remains less than 2%. Such operation may continue for up to 14 days. If there are no monitors OPERABLE, WASTE GAS HOLDUP SYSTEM operation may continue provided a grab sample is taken and analyzed from the onservice gas decay tank once per 4 hours and the oxygen concentration remains less than 1%. With oxygen concentration exceeding 1%, reduce the oxygen concentration to less than 1% within 48 hours, or be in HOT STANDBY within the next 6 hours.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
b. Effluent System Flow Rate Measuring Device	P	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(4)	M	**
b. Oxygen Monitors	D	N.A.	Q(5)	M	**
3. MAIN CONDENSER EVACUATION AND TURBINE GLAND SEALING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
4. REACTOR AUXILIARY BUILDING VENTILATION SYSTEM (PLANT STACK)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release#	D	M	R(3)	Q(6)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
5. FUEL HANDLING BUILDING VENTILATION SYSTEM (NORMAL)					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***
b. Iodine Sampler	W	N.A.	N.A.	N.A.	***
c. Particulate Sampler	W	N.A.	N.A.	N.A.	***
d. Flow Rate Monitor	D	N.A.	R	Q	***
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	***

#Automatic termination of containment purge only.

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

*At all times.

**During WASTE GAS HOLDUP SYSTEM operation.

***When irradiated fuel is in the spent fuel pool.

(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.
3. Instrument indicates a downscale 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 (NBS) 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. Zero 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. Zero volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the alarm/trip setpoint and that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm set.
2. Circuit failure.
3. Instrument controls not set in operate mode.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lead or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required overspeed protection system otherwise inoperable, with 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 31 days by cycling each of the following valves through at least one complete cycle from the running position.

   1. Four high pressure throttle valves.
   2. Four high pressure governor valves.
   3. Six low pressure reheat stop valves.
   4. Six low pressure reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

e. At least once per 40 months by inspecting the low pressure turbine discs.

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*With any main steam isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant Loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

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

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 50% of wide range indication at least once per 12 hours.

*All reactor coolant pumps may be deenergized 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.

**See Special Test Exception 3.10.5.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump,**

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or shutdown cooling 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 shutdown cooling pumps (LPSI pumps) may be deenergized 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.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 285°F unless (1) the pressurizer water volume is less than 900 cubic feet or (2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

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 the secondary side water level to be $\geq$ 50% of wide range indication at least once per 12 hours.

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4 At least two of the loop(s)/trains listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loop shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump**,

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump**,

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 5 with reactor coolant loops filled**.

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required loops to OPERABLE status or to restore the required level as soon as possible.

b. With no reactor coolant or shutdown cooling 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.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 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.

4.4.1.4.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$50% of wide range indication at least once per 12 hours.

4.4.1.4.3 At least one reactor coolant loop or shutdown cooling train shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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*All reactor coolant pumps and shutdown cooling pumps (LPSI pumps) may be deenergized 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.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 285°F unless (1) the pressurizer water volume is less than 900 cubic feet or (2) the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.5 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

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.

b. With no shutdown cooling 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 shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.5 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump (LPSI pump) may be deenergized 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

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODE 4.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes (except cooldown in shutdown cooling) and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with:

a. A steady-state water volume greater than or equal to 26% indicated level (350 cubic feet) but less than or equal to 62.5% indicated level (900 cubic feet), and,

b. At least two groups of pressurizer heaters powered from Class 1E buses each having a nominal capacity of 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required pressurizer heaters OPERABLE, restore at least two groups 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 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

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying the above pressurizer heaters are automatically shed from the emergency power sources upon the injection of an SIAS test signal.

b. Verifying that the above heaters can be manually placed and energized on the emergency power source from the control room without an SIAS test signal present.

3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. The tubes selected for each inservice inpection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4a.8.) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above requ.red inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or main feedwater line break.

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3c., above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4–2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C–1	None	N. A.	N. A.	N. A.	N. A.
	C–2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C–1	None	N. A.	N. A.
			C–2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug defective tubes
					C–3	Perform action for C–3 result of first sample
			C–3	Perform action for C–3 result of first sample	N. A.	N. A.
	C–3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C–1	None	N. A.	N. A.
			Some S. G.s C–2 but no additional S. G. are C–3	Perform action for C–2 result of second sample	N. A.	N. A.
			Additional S. G. is C–3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = \frac{6}{n}\%$ Where n is the number of steam generators inspected during an inspection

REACTOR COOLANT SYSTEM

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level and flow monitoring system, and

c. Either the containment air cooler condensate flow switches on at least three coolers or a 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 and/or particulate radioactivity 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

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. 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,

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

c. Containment air cooler condensate flow switches - performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 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 720 gallons per day through any one steam generator,

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

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia 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 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 one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Commencing an RCS inventory balance within 1 hour to determine the leak rate when RCS leakage is alarmed and confirmed in a flow path with no flow rate indication.

b. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

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

SURVEILLANCE REQUIREMENTS (Continued)

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.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1, Section A and Section B, 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. Following valve actuation for valves in Section B due to automatic or manual action or flow through the valve:

   1. Within 24 hours by verifying valve closure, and

   2. Within 31 days by verifying leakage rate.

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

4.4.5.2.3 Each Reactor Coolant System pressure isolation valve power-operated valve specified in Table 3.4-1, Section C, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months, and

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

SECTION A

SI-329A	SIT Check
SI-329B	"
SI-330A	"
SI-330B	"
SI-336A	Cold Leg Injection Check
SI-336B	"
SI-335A	"
SI-335B	"
SI-510A	Hot Leg Injection Check
SI-512A	"
SI-510B	"
SI-512B	"
SI-241	HPSI Check
SI-242	"
SI-243	"
SI-244	"

SECTION B

SI-142A	LPSI Check
SI-142B	"
SI-143A	"
SI-143B	"

SECTION C POWER-OPERATED VALVES

SI-401A	SDC Suction Isolation
SI-401B	"
SI-405A	"
SI-405B	"

(a) Maximum Allowable Leakage (each valve):

1. Except as noted below, leakage rates greater than 1.0 gpm are unacceptable.

2. For power-operated valves (POVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. For power-operated valves (POVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 200 psid.

3/4.4.6 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psia, if applicable, and 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 operation prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.6 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

MODES 1, 2, 3, 4, and 5:

d. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE

AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 % of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

Until the specific activity of the primary coolant system is restored within its limits.

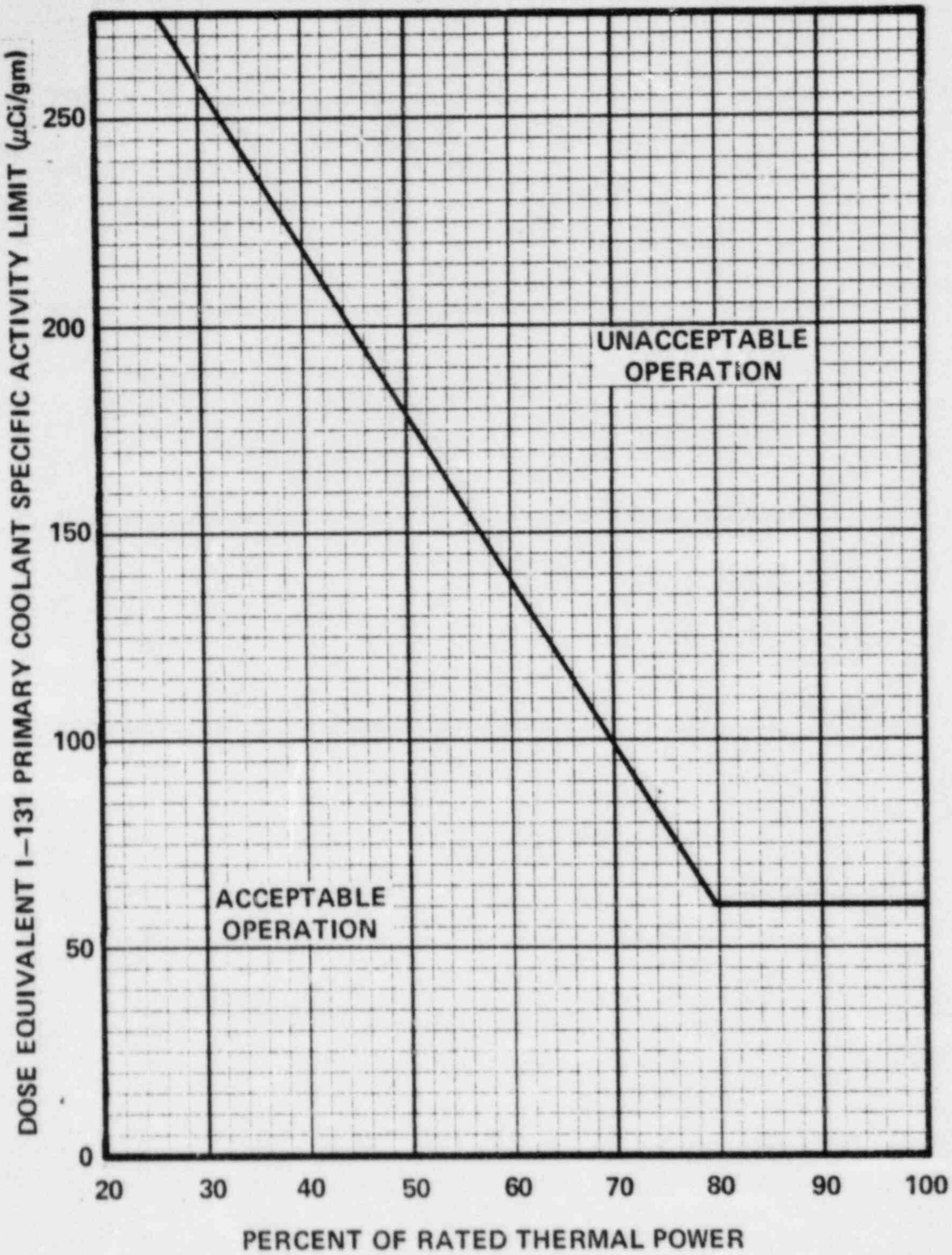

FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 µCi/GRAM DOSE EQUIVALENT I-131

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 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 rate of 30°F per hour with Reactor Coolant System cold leg temperature less than 200°F.

b. A maximum heatup rate of 50°F per hour with Reactor Coolant System cold leg temperature greater than 200°F and less than or equal to 345°F.

c. A maximum heatup rate of 60°F per hour with Reactor Coolant System cold leg temperature greater than 345°F.

d. A maximum cooldown rate of 10°F per hour with Reactor Coolant System cold leg temperature less than 135°F.

e. A maximum cooldown rate of 30°F per hour with Reactor Coolant System cold leg temperature greater than or equal to 135°F and less than or equal to 200°F.

f. A maximum cooldown rate of 100°F per hour with Reactor Coolant System cold leg temperature greater than 200°F.

g. A maximum temperature change of less than or equal to 10°F in any 1-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 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 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.

4.4.8.1.2 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 Part 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. The adjusted reference temperature resulting from neutron irradiation shall be calculated based on the greater of the following:

a. Actual shift in the RT_{NDT} as measured by impact testing of 88114/0145 weld metal;

b. Predicted shift in RT_{NDT} for E8018/BOCA weld metal as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

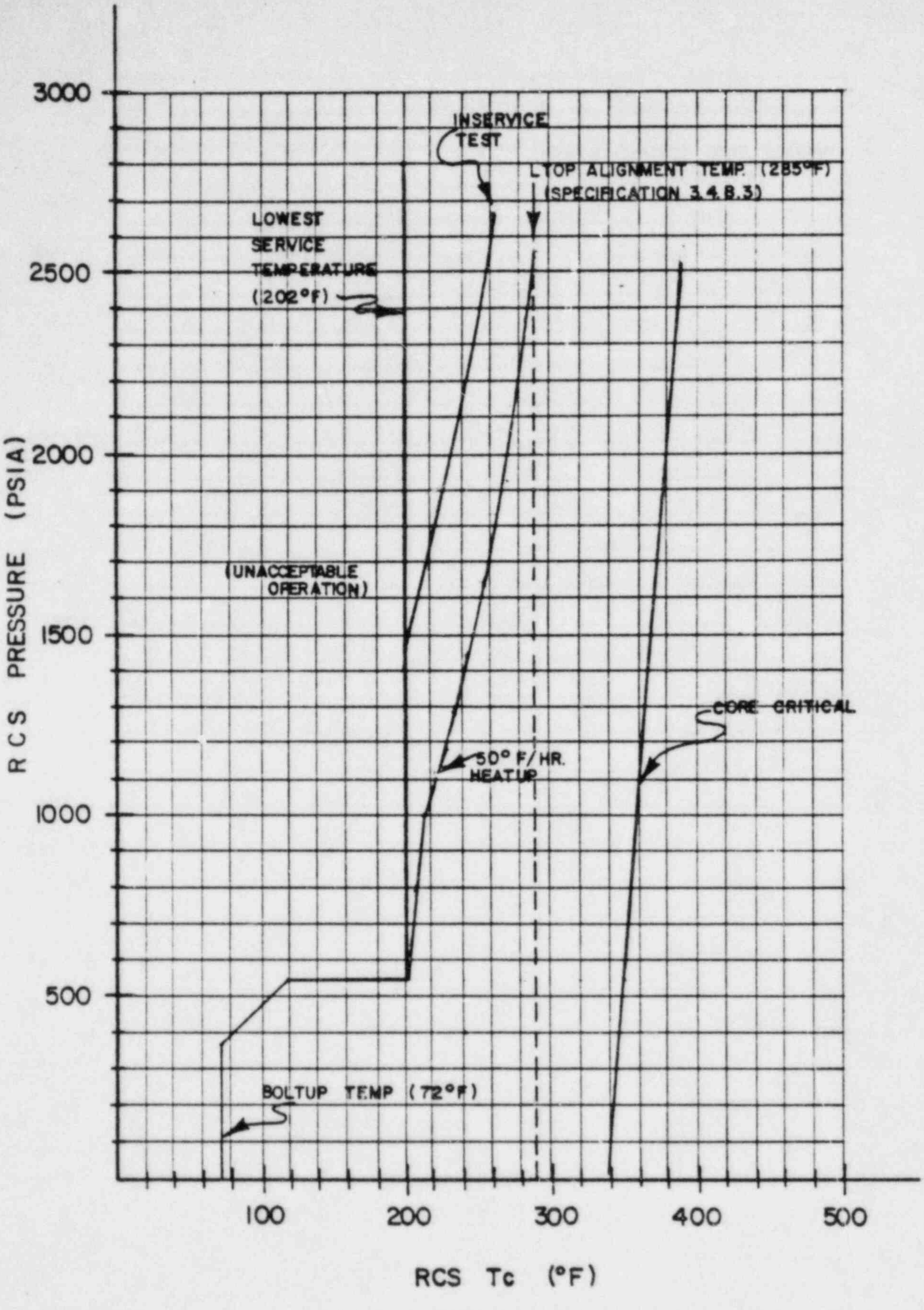

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS
FOR 0-8 EFFECTIVE FULL POWER YEARS (HEATUP)

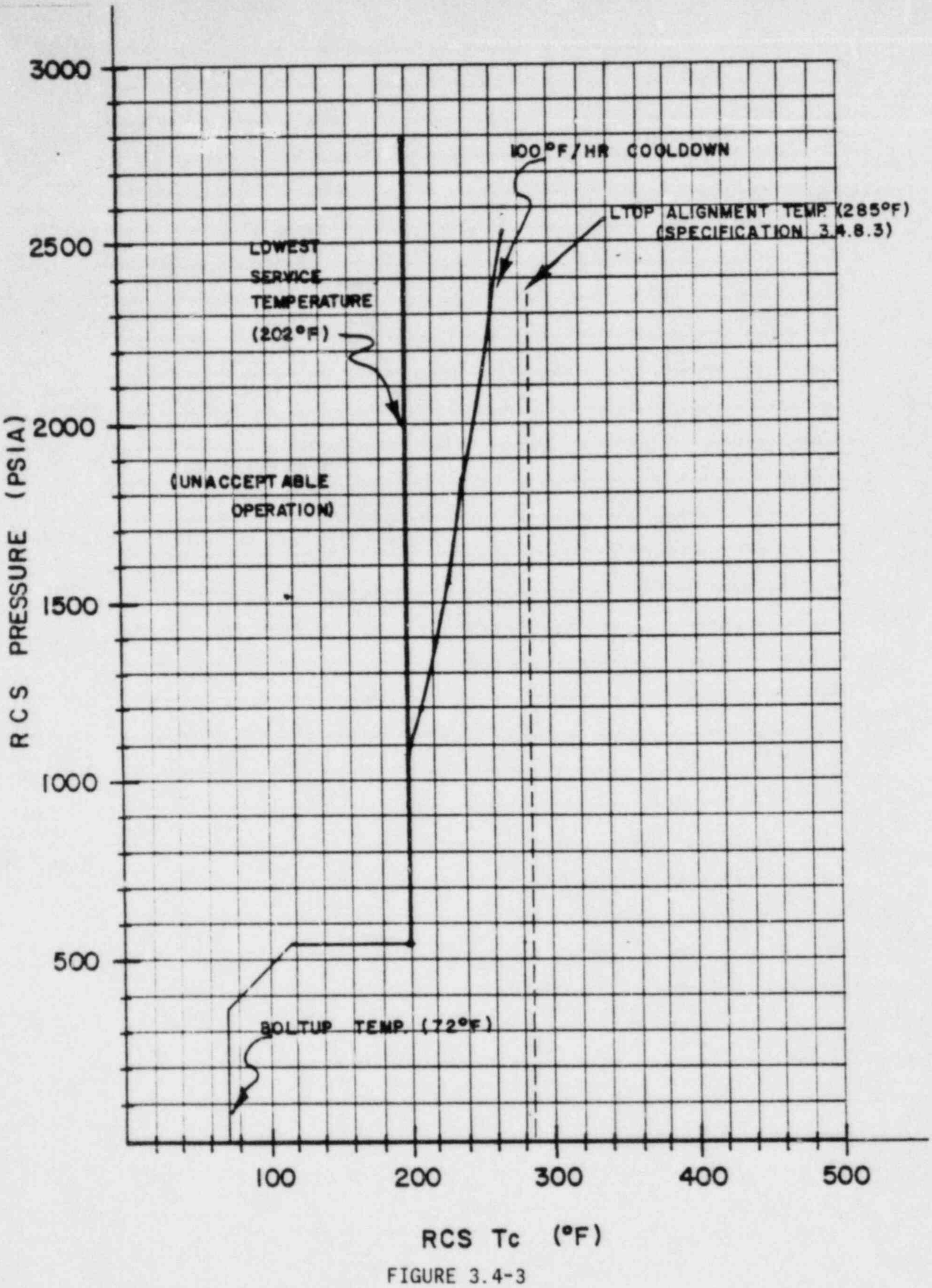

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS
FOR 0-8 EFFECTIVE FULL POWER YEARS (COOLDOWN)

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)*
1	83°	1.50	Standby
2	97°	1.50	4.0 EFPY
3	104°	1.50	11.0 EFPY
4	284°	1.50	18.0 EFPY
5	263°	1.50	Standby
6	277°	1.50	Standby

*Withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

REACTOR COOLANT SYSTEM

PRESSURIZER HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup rate of 200°F per hour,

b. A maximum cooldown rate of 200°F per hour, and

c. A maximum spray nozzle usage factor of 0.65.

APPLICABILITY: At all times.

ACTION:

a. With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

b. With the spray nozzle usage factor > 0.65, comply with requirements of Table 5.7-1.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

4.4.8.2.3 Each spray cycle and the corresponding ΔT (water temperature differential) shall be recorded whenever main spray is initiated with a ΔT (water temperature differential) of > 130°F and whenever auxiliary spray is initiated with a ΔT (water temperature differential) of 140°F.

LIMITING CONDITION FOR OPERATION

3.4.8.3 Low temperature overpressure protection shall be provided by:

a. At least one of the following overpressure protection systems being OPERABLE:

   1. Both OPERABLE Shutdown Cooling (SDC) System suction line relief valves (SI-406A and SI-406B) each with a lift setting of less than or equal to 430 psia aligned to the Reactor Coolant System, or,

   2. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

b. Establishing less than 100°F ΔT between RCS and steam generator temperature or ensuring the pressurizer water volume is less than 900 cubic feet (62.5%), prior to starting any reactor coolant pump.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 285°F, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one Shutdown Cooling System suction line relief valve inoperable, restore the inoperable valve to OPERABLE status within 7 days, or be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

b. With no Shutdown Cooling System suction line relief valves OPERABLE and capable of providing Reactor Coolant System overpressure protection, either:

   1. Restore at least one Shutdown Cooling System suction relief valve to OPERABLE status within 1 hour, or

   2. Be in at least COLD SHUTDOWN and depressurize and vent the RCS within the next 8 hours.

c. In the event either the Shutdown Cooling System suction relief valves(s) or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the Shutdown Cooling System suction relief valve(s) or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each Shutdown Cooling System suction line relief valve shall be demonstrated OPERABLE:

a. By verifying that each valve in the suction path between the Reactor Coolant System and the shutdown cooling relief valve is key-locked open in the control room at least once per 12 hours.

b. At least every 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.2 The RCS vent(s) and all valves in the vent path shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 70°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F, except during hydrostatic testing of components that are nonisolable from the Reactor Coolant System, then restore the structural integrity prior to increasing the Reactor Coolant System temperature more than 30°F above the minimum temperature required by NDT considerations.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 At least one Reactor Coolant System vent path consisting of at least two valves in series powered from emergency busses shall be OPERABLE and closed at each of the following locations:

a. Reactor vessel head, and

b. Pressurizer steam space.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more Reactor Coolant System vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent valve through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

c. Verifying flow through the Reactor Coolant System vent paths during venting during COLD SHUTDOWN or REFUELING.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1679 (78%) and 1807 (83.8%) cubic feet,

c. Between 1720 and 2300 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

APPLICABILITY: MODES 1, 2, 3*, and 4*.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 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 safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each safety injection tank isolation valve is open.

---

*With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 1332 (60%) and 1807 (83.8%) cubic feet. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 625 psig, a boron concentration of between 1720 and 2300 ppm boron, and a contained borated water volume of between 888 (39%) and 1807 (83.8%) cubic feet. In MODE 4 with pressurizer pressure less than 392 psia (700 psia for remote shutdown from LCP-43), the safety injection tanks may be isolated.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days when the RCS pressure is above 1750 psia, by verifying that the isolation valve operator breakers are padlocked in the open position.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. When an actual or simulated RCS pressure signal exceeds 535 psia, and

   2. Upon receipt of a safety injection test signal.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent emergency core cooling system (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water storage pool on a safety injection actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem 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. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1750 psia.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
a. 2SI-V1556 (SI-506A)	a. Hot Leg Injection	a. SHUT
b. 2SI-V1557 (SI-502A)	b. Hot Leg Injection	b. SHUT
c. 2SI-V1558 (SI-502B)	c. Hot Leg Injection	c. SHUT
d. 2SI-V1559 (SI-506B)	d. Hot Leg Injection	d. SHUT

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. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the safety injection system sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure (actual or simulated) is 700 ± 20 psia.

2. A visual inspection of the safety injection system sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 97.5 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 4 ± 0.01 grams of TSP from a TSP storage basket is submerged, without agitation, in 4 ± 0.1 liters of 120 ± 10 °F water borated within RWSP boron concentration limits, the pH of the mixed solution is raised to greater than or equal to 7 within 3 hours.

5. A visual inspection of the TSP storage baskets for evidence of aggregation and the mechanical dispersal of any aggregations found.

e. 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 SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

    a. High pressure safety injection pump.

    b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the low pressure safety injection pumps stop, the safety injection system sump isolation valves open.

f. By verifying that each of the following pumps required to be OPERABLE performs as indicated on recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pumps develop a total head of greater than or equal to 1400 psid for pump A, 1431 psid for pump B and 1429 psid for pump A/B.

2. Low pressure safety injection pump discharge pressure greater than or equal to 177 psig.

SURVEILLANCE REQUIREMENTS (Continued)

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves by verifying that each ECCS throttle valve opens to the proper throttled position each time the valve is cycled:

HPSI System Valve Number				LPSI System Valve Number	
a.	SI-225A	e.	SI-227A	a.	SI-138A
b.	SI-225B	f.	SI-227B	b.	SI-138B
c.	SI-226A	g.	SI-228A	c.	SI-139A
d.	SI-226B	h.	SI-228B	d.	SI-139B

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow characteristics:

HPSI System - Single Pump (Cold leg injection mode)

The sum of the injection lines flow rates, excluding the highest flow rate, is greater than or equal to 658 gpm for HPSI Pump A running, 665 gpm for HPSI Pump B running, and 650 gpm for HPSI Pump A/B running, with a maximum differential pressure of less than or equal to 528 psid for HPSI Pump A, 472 psid for HPSI Pump B, and 489 psid for HPSI Pump A/B.

HPSI SYSTEM - Single Pump (Hot/cold leg injection mode)

With the system operating in the hot/cold leg injection mode, the hot leg flow must be greater than or equal to 436 gpm and within ± 10% of the cold leg flow.

LPSI System - Single Pump

Flow for each pump is greater than or equal to 4810 with the total developed head greater than or equal to 268 feet but less than or equal to 292 feet.

SURVEILLANCE REQUIREMENTS (Continued)

i. Each time HPSI Pump A/B is placed in or taken out of service in place of HPSI Pump A or B, the pump being placed in service shall be demonstrated OPERABLE by:

1. Verifying that each valve in the flow path is in its correct position; and

2. Verifying the pump starts manually and upon receipt of a SIAS test signal; and

3. Performing Surveillance Requirement 4.5.2f.1., if not previously accomplished within the required frequency.

j. Following any maintenance which drains portions of the system, by venting the ECCS pump casings and discharge piping high points.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water storage pool on a safety injection actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

*With pressurizer pressure less than 1750 psia.

3/4.5.4 REFUELING WATER STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage pool shall be OPERABLE with:

a. A minimum contained borated water volume of 475,500 gallons (82% indicated level),

b. Between 1720 and 2300 ppm of boron, and

c. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the refueling water storage pool inoperable, restore the pool to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWSP shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWSP temperature when the RAB air temperature is less than 55°F or greater than 100°F.

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 1 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 in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a, 44 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the 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.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 44 psig, or

2. Less than or equal to L_t, 0.25 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 22 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests as identified in Table 3.6-1, when pressurized to P_a.

c. A combined bypass leakage rate of less than or equal to 0.06 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.06 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 L_a, and the bypass leakage rate to less than or equal to 0.06 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during

SURVEILLANCE REQUIREMENTS (Continued)

shutdown at either P_a, 44 psig, or at P_t, 22 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1. Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, are equal to or less than 0.25 L_a.

   2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a, 44 psig, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.7.2.

SURVEILLANCE REQUIREMENTS (Continued)

f. The combined bypass leakage rate shall be determined to be less than or equal to 0.06 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 44 psig, during each Type A test.

g. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

h. The provisions of Specification 4.0.2 are not applicable.

TABLE 3.6-1

CONTAINMENT LEAKAGE PATHS

PENETRATION NO.	SYSTEM NAME	VALVE TAG NO.	TEST TYPE
7	Demineralized Water	2DW-V609A/B (PMU 151) 2DW-V610 (PMU 152)	Bypass/Type C
8	Station Air	2SA-V610A/B (SA 908) 2SA-V602A/B (SA 909)	Bypass/Type C
9	Instrument Air	2IA-F601A/B (IA 909) 2IA-V602A/B (IA 910)	Bypass/Type C
10	Containment Purge Inlet	2HV-B151A (CAP 103) 2HV-B152A (CAP 104)	Type C
11	Containment Purge Exhaust	2HV-B154B (CAP 204) 2HV-B153B (CAP 203)	Type C
12	Containment Vacuum Relief	2HV-B157B (CVR 101) 2HV-B181B (CVR 102	Type C
13	Containment Vacuum Relief	2HV-B156A (CVR 201) 2HV-V181B (CVR 202)	Type C
14	Nitrogen Systems Supply to Reactor Bldg	2NG-F604 (NG 157) 2NG-V666 (NG 158)	Bypass/Type C
*23	CCW to RCPs and CEDM Cooler	2CC-F146A/B (CC-641) 2CC-V242A/B (CC-644)	Type C
*24	CCW to RCPs and CEDM Cooler	2CC-F147A/B (CC-713) 2CC-F243A/B (CC-710)	Type C
25	Fuel Transfer Containment & Fuel Handling Building		Bypass/Type B

*These penetrations shall be tested prior to STARTUP following first refueling outage.

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

PENETRATION NO.	SYSTEM NAME	VALVE TAG NO.		TEST TYPE
26	Chemical & Volume Control Letdown Line	2CH-F1518A/B 1CH-F2501A/B	(CVC 109) (CVC 103)	Bypass/Type C
28	Sampling Line from Reactor Coolant Line	2SL-F1504A/B 2SL-F1501A/B	(PSL 107) (PSL 105)	Bypass/Type C
29	Sampling Line from Pressurizer Surge Line	2SL-F1505A/B 2SL-F1502A/B	(PSL 204) (PSL 203)	Bypass/Type C
30	Sampling Line from Pressurizer Steam Space	2SL-F1506A/B 2SL-F1503A/B	(PSL 304) (PSL 303)	Bypass/Type C
31	Waste Management from Containment Vent Header	2WM-F158A/B 2WM-F157A/B	(GWM 105) (GWM 104)	Bypass/Type C
42	Containment Sump Pump Discharge/Post Accident Sample Return	2WM-F105A/B 2WM-F104A/B	(SP 106) (SP 105)	Bypass Type C
43	Boron Management Reactor Drain Tank Outlet	2BM-F109A/B 2BM-F108A/B	(BM 110) (BM 109)	Bypass/Type C
44	Chemical & Volume Control from Reactor Pump Controlled Bleedoff	2CH-F1512A/B 2CH-F1513A/B	(CVC 401) (RC 606)	Bypass/Type C
45	CARS Makeup to Containment	2HV-B187B 2HV-V185B	(CAR 101B) (CAR 102B)	Bypass/Type C
46	CARS Makeup to Containment	2HV-B188A 2HV-V184A	(CAR 101A) (CAR 102A)	Bypass/Type C
47	CARS Exhaust from Containment	2HV-B192B 2HV-F254B	(CAR 202B) (CAR 201B)	Bypass/Type C

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

PENETRATION NO.	SYSTEM NAME	VALVE TAG NO.		TEST TYPE
48	CARS Exhaust from Containment	2HV-B190A 2HV-F253A	(CAR 202A) (CAR 201A)	Bypass/Type C
49	Containment Atmosphere Monitoring Inlet and Outlet	2CA-E605A 2CA-E604B 2CA-V607 2CA-E606A	(ARM 110) (ARM 109) (ARM 104) (ARM 103)	Type C
51	Refueling Cavity Purification Inlet	2FS-V145A/B 2FS-V144A/B	(FS 405) (FS 406)	Bypass/Type C
59	Safety Injection System from SI Tank to Refueling Water Storage Pool	2SI-V1570 2SI-V1561A/B	(SI 344) (SI 343)	Bypass/Type C
60	Fire Protection System to Reactor Building	2FP-F127 2FP-V128	(FP 601A) (FP 602A)	Bypass/Type C
61	Fire Protection System to Reactor Building	2FP-F129 2FP-V130	(FP 601B) (FP 602B)	Bypass/Type C
62	Water from Refueling Cavity to RWSP	2FS-V165A/B 2FS-V164A/B	(FS 416) (FS 415)	Bypass/Type C
63	Containment Leakage Rate Test Connection	2SA-V114 Blind Flange	(LRT 109) NA	Type C
65	Containment Leakage Rate Test Connection and Instrument H&V	2SA-V609 2SA-V611	(LRT 202) (LRT 204)	Type C
66	Hydrogen Analyzer Supply and Return	2HA-E609A 2HA-E608A 2HA-E610A 2HA-F637A	(HRA 110A) (HRA 109A) (HRA 126A) (HRA 128A)	Type C

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

PENETRATION NO.	SYSTEM NAME	VALVE TAG NO.		TEST TYPE
67	Hydrogen Analyzer Supply and Return	2HA-E629B 2HA-E628B 2HA-E630B 2HA-E638B	(HRA 110B) (HRA 109B) (HRA 126B) (HRA 128B)	Type C
71	Demineralized Water	2DW-V642 2DW-V643	(CMU 244) (CMU 245)	Bypass/Type C
Escape Lock	NA	None		Bypass/Type B
Personnel Lock	NA	None		Type B
Electrical Penetrations	NA	All Primary Canisters except welded spares		Type B
Equipment Hatch	NA	None		Type B
Expansion Bellows 1, 2, 3, 4, 25, 32, 33, 43	Various	None		Type B

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 44 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock 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

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage to be less than or equal to 0.005 L_a when determined by pressure decay or precision flow rate measurement with the volume between the door seals pressurized to greater than or equal to 10 psig, for at least 15 minutes,

b. By conducting overall air lock leakage tests at not less than P_a, 44 psig, and verifying the overall air lock leakage rate is within its limit:

1. At least once per 6 months,# and

2. Prior to establishing CONTAINMENT INTEGRITY 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

*This constitutes an exemption to Appendix J of 10 CFR Part 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained within the limits of Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

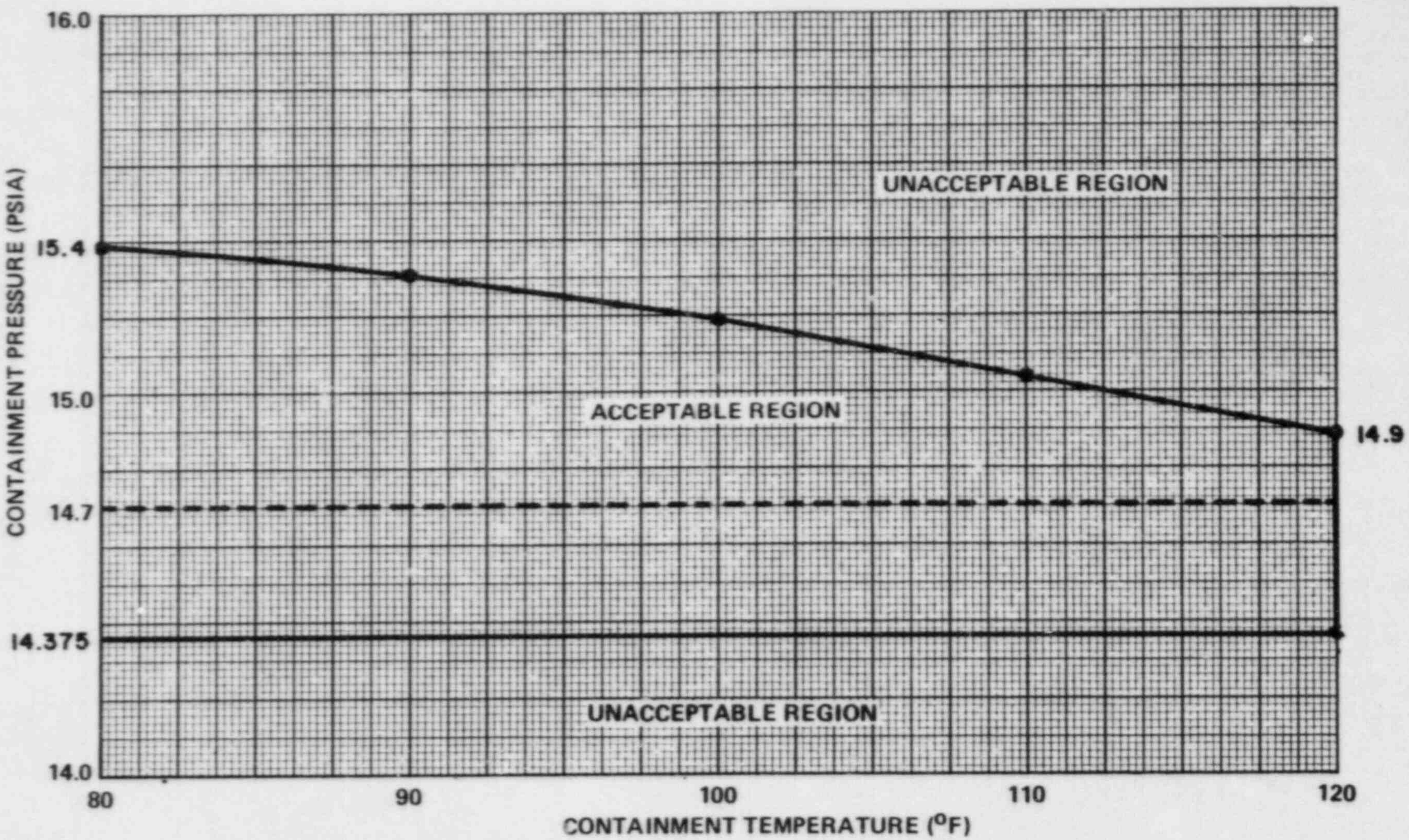

FIGURE 3.6-1

CONTAINMENT PRESSURE VS TEMPERATURE

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120 °F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120 °F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any three of the following locations and shall be determined at least once per 24 hours:

Location

a. Containment Fan Cooler No. 1A Air Intake

b. Containment Fan Cooler No. 1B Air Intake

c. Containment Fan Cooler No. 1C Air Intake

d. Containment Fan Cooler No. 1D Air Intake

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 15 days.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve (CAP 103, CAP 104, CAP 203, and CAP 204) shall be OPERABLE and may be open at no greater than the 52° open position allowed by the mechanical stop for less than 90 hours per 365 days.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a containment purge supply and/or exhaust isolation valve(s) open for greater than or equal to 90 hours per 365 days at any open position, close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirement 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The cumulative time that the purge supply or exhaust isolation valves are open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.2 At least once per 3 months* each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.06 L_a when pressurized to P_a.

4.6.1.7.3 Each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE during each COLD SHUTDOWN exceeding 24 hours by verifying that the mechanical stops limit the valve opening to a position $\leq$ 52° open.

*Until STARTUP following the first refueling outage, the containment purge supply and exhaust isolation valves shall be tested during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 92 days.

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 RWSP on a containment spray actuation signal and automatically transferring suction to the safety injection system sump on a recirculation actuation signal. Each spray system flow path from the safety injection system sump shall be via an OPERABLE shutdown cooling heat exchanger.

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 at least 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 following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the water level in the containment spray header riser is $\geq$ 149.5 feet MSL elevation.

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 correctly positioned to take suction from the RWSP.

c. By verifying, that on recirculation flow, each pump develops a total head of greater than or equal to 219 psid when tested pursuant to Specification 4.0.5.

d. 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 CSAS test signal.

---

*When shutdown cooling is in operation, no independent containment spray systems are required to be OPERABLE.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that upon a recirculation actuation test signal, the safety injection system sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

3. Verifying that each spray pump starts automatically on a CSAS test signal.

e. 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

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Two independent groups of containment cooling fans shall be OPERABLE with two fan systems to each group.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one group of the above required containment cooling fans inoperable, restore the inoperable group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable group of containment cooling fans to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each fan group not already running from the control room and verifying that each fan group operates for at least 15 minutes.

   2. Verifying a cooling water flow rate of greater than or equal to 625 gpm to each cooler.

b. At least once per 18 months by:

   1. Verifying that each fan group starts automatically on an SIAS test signal.

   2. Verifying a cooling water flow rate of greater than or equal to 1325 gpm to each cooler.

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve specified in Table 3.6-2 shall be demonstrated OPERABLE at least once per 18 months by:

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

b. Verifying that on a containment Radiation-High test signal, each containment purge 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.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Containment Isolation (CIAS)			
1	2MS-V670 (MS-120A)	Main Steam Drain	10
1	2MS-V671 (MS-119A)	Main Steam Drain	10
2	2MS-V663 (MS-120B)	Main Steam Drain	10
2	2MS-V664 (MS-119B)	Main Steam Drain	10
5	2BD-F603 (BD102A)	Steam Generator Blowdown	10
5	2BD-F604 (BD103A)	Steam Generator Blowdown	10
6	2BD-F605 (BD102B)	Steam Generator Blowdown	10
6	2BD-F606 (BD103B)	Steam Generator Blowdown	10
9	2IA-F601A/B (IA909)	Instrument Air	5
14	2NG-F604 (NG157)	Nitrogen Supply	5
26	2CH-F1518A/B (CVC109)	CVCS Letdown	10
26	1CH-F2501A/B (CVC103)	CVCS Letdown	10
28	2SL-F1504A/B (PSL107)	RCS Sample	10
28	2SL-F1501A/B (PSL105)	RCS Sample	10
29	2SL-F1505A/B (PSL204)	Pressurizer Surge Line Sample	10

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Containment Isolation (Continued)			
29	2SL-F1502A/B (PSL203)	Pressurizer Surge Line Sample	10
30	2SL-F1506A/B (PSL304)	Pressurizer Sample	10
30	2SL-F1503A/B (PSL303)	Pressurizer Sample	10
31	2WM-F158A/B (GWM105)	Waste Gas Vent Header	7
31	2WM-F157A/B (GWM104)	Waste Gas Vent Header	7
33	2SI-E655 (SI-6011)	SIS Sampling	5
33	2SI-E654 (SI-6012)	SIS Sampling	5
42	2WM-F105A/B (SP106)	Containment Sump Pump Discharge	7
42	2WM-F104A/B (SP105)	Containment Sump Pump Discharge	7
43	2BM-F109A/B (BM110)	Reactor Drain Tank Outlet	7
43	2BM-F108A/B (BM109)	Reactor Drain Tank Outlet	7
44	2CH-F1512A/B (CVC401)	RCP Bleedoff	10
44	2CH-F1513A/B (RC606)	RCP Bleedoff	10
47	2HV-F254B (CAR201B)	CARS Exhaust	10
48	2HV-F253A (CAR201A)	CARS Exhaust	10
49	2CA-E605A (ARM110)	Containment Atmosphere Monitor	5

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Containment Isolation (Continued)			
49	2CA-E604B (ARM109)	Containment Atmosphere Monitor	5
49	2CA-E606E (ARM103)	Containment Atmosphere Monitor	5
52	2SL-F602 (PSL406A)	Steam Generator Blowdown Sample	10
52	2SL-F601 (PSL404A)	Steam Generator Blowdown Sample	10
53	2HV-E634A (CVR401A)	Containment Vacuum Relief Instrument Line	5
59	2SI-F1561A/B (SI343)	SIT Drain to RWSP	10
60	2FP-F127 (FP601A)	Containment Fire Water Header	10
61	2FP-F129 (FP601B)	Containment Fire Water Header	10
65	2HV-E633B (CVR401B)	Containment Vacuum Relief Instrument Line	5
66	2HA-E609A (HRA110A)	Hydrogen Analyzer	5
66	2HA-E608A (HRA109A)	Hydrogen Analyzer	5
66	2HA-E610A (HRA126A)	Hydrogen Analyzer	5
67	2HA-E629B (HRA110B)	Hydrogen Analyzer	5
67	2HA-E628B (HRA109B)	Hydrogen Analyzer	5
67	2HA-E630B (HRA126B)	Hydrogen Analyzer	5

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Containment Isolation (Continued)			
68	2SL-F604 (PSL406B)	Steam Generator Blowdown Sample	10
68	2SL-F603 (PSL404B)	Steam Generator Blowdown Sample	10
2. Containment Purge (CIAS/CPIS)			
10	2HV-B151A (CAP103)	Containment Purge Inlet	5
10	2HV-B152A (CAP104)	Containment Purge Inlet	5
11	2HV-B154B (CAP204)	Containment Purge Outlet	5
11	2HV-B153B (CAP203)	Containment Purge Outlet	5
3. Safety Injection Actuation Signal (SIAS)			
26	1CH-F2501A/B(CVC103)	CVCS Letdown	10
32	2SI-L101A (SI 602A)	SI from SIS Sump	N.A.
33	2SI-L102B (SI 602B)	SI from SIS Sump	N.A.
4. Main Steam Isolation Signal (MSIS)			
1	2MS-V602A (MS 124A)	Main Steam	N.A.
1	2MS-F714 (SSL 301A)	Main Steam Sample	N.A.
2	2MS-V604B (MS 124B)	Main Steam	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
4. Main Steam Isolation Signal (MSIS) (Continued)			
2	2MS-F715 (SSL 301B)	Main Steam Sample	N.A.
3	2FW-V823A (FW 184A)	Main Feedwater	N.A.
3	2FW-V847B (EFW 229A)	Emergency Feedwater	N.A.
3	2FW-V848A (EFW 228A)	Emergency Feedwater	N.A.
3	2FW-V851B (EFW 224A)	Emergency Feedwater	N.A.
3	2FW-V852A (EFW 223A)	Emergency Feedwater	N.A.
4	2FW-V824B (FW 184B)	Main Feedwater	N.A.
4	2FW-V849A (EFW 229B)	Emergency Feedwater	N.A.
4	2FW-V850B (EFW 228B)	Emergency Feedwater	N.A.
4	2FW-V853A (EFW 224B)	Emergency Feedwater	N.A.
4	2FW-V854B (EFW 223B)	Emergency Feedwater	N.A.
5. Contain Spray Actuation Signal (CSAS)			
23	2CC-F146A/B (CC 641)	CCW to RCPs and CEDM Cooler	5
24	2CC-F147A/B (CC 713)	CCW from RCPs and CEDM Cooler	5
24	2CC-F243A/B (CC 710)	CCW from RCPs and CEDM Cooler	5

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

6. Manual/Remote Manual

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1	2MS-V768 (MS1244A)*	MSIV Bypass	N.A.
1	2MS-V697 (NG412A)	Main Steam N_2 Blanket	N.A.
1	2MS-V611A (MS401A)*	Steam to Emergency Steam Generator Feed Pump Turbine	N.A.
1	2MS-PM629A (MS116A)*	Atmospheric Steam Dump	N.A.
2	2MS-V698 (NG412B)	Main Steam N_2 Blanket	N.A.
2	2MS-V612B (MS401B)*	Steam to Emergency Steam Generator Feed Pump Turbine	N.A.
2	2MS-PM630B (MS116B)*	Atmospheric Steam Dump	N.A.
2	2MS-V710B (MS1244B)*	MSIV Bypass	N.A.
7	2DW-V609A/B (PMU151)*	Demineralized Water	N.A.
8	2SA-V601A/B (SA908)*	Station Air	N.A.
12	2HV-B157B (CVR 101)*	Vacuum Relief	N.A.
13	2HV-B156A (CVR 201)*	Vacuum Relief	N.A.
15	2CC-F157B2 (CC 807B)*	CCW to Containment Fan Cooler Units	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
6. Manual/Remote Manual (Continued)			
16	2CC-F161B2 (CC 823B)*	CCW from Containment Fan Cooler Units	N.A.
17	2CC-F158A1 (CC 823A)*	CCW from Containment Fan Cooler Units	N.A.
18	2CC-F154A1 (CC 807A)*	CCW to Containment Fan Cooler Units	N.A.
19	2CC-F159A2 (CC 822A)*	CCW from Containment Fan Cooler Units	N.A.
20	2CC-F155A2 (CC 808A)*	CCW to Containment Fan Cooler Units	N.A.
21	2CC-F156B1 (CC 808B)*	CCW to Containment Fan Cooler Units	N.A.
22	2CC-F160B1 (CC 822B)*	CCW from Containment Fan Cooler Units	N.A.
27	2CH-F1529A/B (CVC 209)*	CVCS Charging Line	N.A.
27	1CH-F2505A (CVC 216A)*	CVCS Auxiliary Spray	N.A.
27	1CH-F2505B (CVC 216B)*	CVCS Auxiliary Spray	N.A.
27	1CH-F2504B (CVC 218B)*	CVCS Charging Line	N.A.
27	1CH-F2503A (CVC 218A)*	CVCS Charging Line	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
6. Manual/Remote Manual (Continued)			
34A&B	2CS-F305A (CS 125A)*	Containment Spray	N.A.
34A&B	2CS-E608A (CS 129A)*	Containment Spray	N.A.
35A&B	2CS-F306B (CS 125B)*	Containment Spray	N.A.
35A&B	2CS-E609B (CS 129B)*	Containment Spray	N.A.
36	2SI-V1549A1 (SI 139B)*	SI from LPSI Pumps	N.A.
37	2SI-V1539B1 (SI 138B)*	SI from LPSI Pumps	N.A.
38	2SI-V1541A2 (SI 139A)*	SI from LPSI Pumps	N.A.
39	2SI-V1543B2 (SI 138A)*	SI from LPSI Pumps	N.A.
40	2SI-V326B (SI407B)*	Shutdown Cooling	N.A.
40	1SI-V1501B (SI405B)*	Shutdown Cooling	N.A.
41	2SI-V327A (SI407A)*	Shutdown Cooling	N.A.
41	1SI-V1503A (SI405A)*	Shutdown Cooling	N.A.
45	2HV-B187B (CAR101B)*	CARS Makeup	N.A.
46	2HV-B188A (CAR101A)*	CARS Makeup	N.A.
47	2HV-B192B (CAR202B)*	CARS Exhaust	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
6. Manual/Remote Manual (Continued)			
48	2HV-B190A (CAR202A)*	CARS Exhaust	N.A.
51	2FS-V145A/B (FS405)*	Refueling Cavity Purification Inlet	N.A.
51	2FS-V144A/B (FS406)*	Refueling Cavity Purification Inlet	N.A.
53	2CA-V600 (CVR 301B)*	Instrument H&V	N.A.
55	2SI-V1550A1 (SI 225A)*	SIS from HPSI Loop 1A	N.A.
55	2SI-V1545B1 (SI 225B)*	SIS from HPSI Loop 1A	N.A.
56	2SI-V1546A2 (SI 226A)*	SIS from HPSI Loop 1B	N.A.
56	2SI-V1540B2 (SI 226B)*	SIS from HPSI Loop 1B	N.A.
57	2SI-V1542A3 (SI 227A)*	SIS from HPSI Loop 2A	N.A.
57	2SI-V1547B3 (SI 227B)*	SIS from HPSI Loop 2A	N.A.
58	2SI-V1548A4 (SI 228A)*	SIS from HPSI Loop 2B	N.A.
58	2SI-V1544B4 (SI 228B)*	SIS from HPSI Loop 2B	N.A.
59	2SI-V1570 (SI344)*	SIT Drain to RWSP	N.A.
62	2FS-V165A/B (FS416)	Refueling Cavity Drain	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
6. Manual/Remote Manual (Continued)			
62	2FS-V164A/B (FS415)	Refueling Cavity Drain	N.A.
63	2SA-V114 (LRT101)	ILRT Connection	N.A.
63	2SA-V504 (LRT102)	ILRT Connection	N.A.
65	2SA-V609 (LRT202)	ILRT Test Connection	N.A.
65	2SA-V611 (LRT204)	ILRT Test Connection	N.A.
65	2SA-V610 (LRT201)	ILRT Test Connection	N.A.
65	2SA-V612 (LRT203)	ILRT Test Connection	N.A.
65	2SA-V620 (LRT2011)	ILRT Test Connection	N.A.
65	2SA-V621 (LRT 2031)	ILRT Test Connection	N.A.
65	2CA-V601 (CVR 301B)*	Instrument H&V	N.A.
69	2SI-V1556 (SI 506A)*	SI Hot Leg Injection	N.A.
70	2SI-V1559 (SI 506B)*	SI Hot Leg Injection	N.A.
71	2DW-V642 (CMU244)*	Demineralized Water	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
7. Other			
1	2NG-V621-1 (NG 411A)	For Steam Generator Nitrogen Blanket	N.A.
2	2NG-621-2 (NG 411B)	For Steam Generator Nitrogen Blanket	N.A.
7	2DW-V610 (PMU162)	Demineralized Water Check Valve	N.A.
8	2SA-V602A/B (SA909)	Station Air Check Valve	N.A.
9	2IA-V602A/B (IA910)	Instrument Air Check Valve	N.A.
12	2HV-B181B (CVR 102)	Vacuum Relief	N.A.
13	2HV-B180A (CVR 202)	Vacuum Relief	N.A.
14	2NG-V666 (NG158)	Containment N_2 Supply Check Valve	N.A.
23	2CC-V242A/B (CC644)	CCW to RCPS and CEDM Cooler Check Valve	N.A.
27	1CH-V2506 (CVC 219)	CVCS Charging Line	N.A.
34A&B	2CS-V103A (CS 128A)	Containment Spray	N.A.
35A&B	2CS-V104B (CS 128B)	Containment Spray	N.A.
36	1SI-V1517RL1A (SI 143B)	SI from LPSI Pumps	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
7. Other (Continued)			
37	1SI-V1518RL1B (SI 142B)	SI from LPSI Pumps	N.A.
38	1SI-V1519RL2A (SI 143A)	SI from LPSI Pumps	N.A.
39	1SI-V1520RL2B (SI 142A)	SI from LPSI Pumps	N.A.
45	2HV-V185B (CAR102B)	CARS Makeup Check Valve	N.A.
46	2HV-V184A (CAR102A)	CARS Makeup Check Valve	N.A.
49	2CA-V607 (ARM104)	Containment Atmosphere Monitor Check Valve	N.A.
53	#3401	Containment Vacuum Relief Instrument Line Excess Flow Check Valve	N.A.
55	1SI-V1522RL1A (SI 241)	SIS from HPSI Loop 1A	N.A.
56	1SI-V1523RL1B (SI 242)	SIS from HPSI Loop 1B	N.A.
57	1SI-V1524RL2A (SI 243)	SIS from HPSI Loop 2A	N.A.
58	1SI-V1525RL2B (SI 244)	SIS from HPSI Loop 2B	N.A.
60	2FP-V128 (FP602A)	Containment Fire Water Header Check Valve	N.A.
61	2FP-V130 (FP602B)	Containment Fire Water Header Check Valve	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES**

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
7. Other (Continued)			
65	#3401	Containment Vacuum Relief Excess Flow Check Valve	N.A.
66	2HA-E637A (HRA128A)	Hydrogen Analyzer Check Valve	N.A.
67	2HA-E638B (HRA128B)	Hydrogen Analyzer Check Valve	N.A.
69	1SI-V2506 (SI 510A)	SI Hot Leg Injection	N.A.
70	1SI-V2508 (SI 510B)	SI Hot Leg Injection	N.A.
71	2DW-V643 (CMU245)	Demineralized Water Check Valve	N.A.

*May be opened on an intermittent basis under administrative control.

**The provisions of Specification 3.0.4 are not applicable.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With both containment hydrogen analyzers inoperable, restore at least one analyzer to OPERABLE status within 72 hours and comply with the requirements of ACTION a, or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE by the performance of 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 gases containing:

a. Zero volume percent hydrogen, balance nitrogen.

b. 9.5 volume percent hydrogen, balance nitrogen.

ELECTRIC HYDROGEN RECOMBINERS - W

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.).

   3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5 The primary containment to annulus vacuum relief valves shall be OPERABLE with an actuation setpoint of less than or equal to 0.307 psid (8.5 inches H_2O).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one primary containment to annulus vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be ir at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent shield building ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one shield building 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.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each shield building ventilation system 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 filters and charcoal adsorbers and verifying that the system operates for at least 10 hours continuous 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:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the ventilation 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, and the system flow rate is 10,000 cfm $\pm$ 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 10,000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by 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, Table 2, page 7, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.8 inches water gauge while operating the system at a flow rate of 10,000 cfm $\pm$ 10%.

2. Verifying that the system starts on a safety injection actuation test signal.

3. Verifying that the filter cooling bypass valves can be manually cycled.

4. Verifying that each system produces a negative pressure of greater than or equal to 0.25 inch water gauge in the annulus within 1 minute after a start signal.

5. Verifying that the heaters dissipate 60 $\pm$ 3.0 kW when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 10,000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 10,000 cfm ± 10%.

SHIELD BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.2 SHIELD BUILDING INTEGRITY shall be maintained with an annulus negative pressure greater than 5 inches water gauge.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY 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

4.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated:

a. At least once per 12 hours by verifying the annulus pressure to be within its limits.

b. At least once per 31 days by verifying that each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed.

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.3 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.6.3.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION

With the structural integrity of the shield building not conforming to the above requirements, restore the structural integrity to within the limits 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

4.6.6.3 The structural integrity of the shield building shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. Any abnormal degradation of the shield building detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Linear Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING (± 1%)*	ORIFICE SIZE
	Line No. 1	Line No. 2		
a.	2MS-R613A (MS-106A)	2MS-R619B (MS-106B)	1070 psig	26 in^2
b.	2MS-R614A (MS-108A)	2MS-R620B (MS-108B)	1085 psig	26 in^2
c.	2MS-R615A (MS-110A)	2MS-R621B (MS-110B)	1100 psig	26 in^2
d.	2MS-R616A (MS-112A)	2MS-R622B (MS-112B)	1115 psig	26 in^2
e.	2MS-R617A (MS-113A)	2MS-R623B (MS-113B)	1125 psig	26 in^2
f.	2MS-R618A (MS-114A)	2MS-R624B (MS-114B)	1135 psig	26 in^2

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

TABLE 3.7-2

MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT (PERCENT OF RATED THERMAL POWER)
1	86.8
2	69.4
3	52.1
4	34.7

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency 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 emergency feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying on a STAGGERED TEST BASIS that each motor-driven pump develops a discharge pressure of greater than or equal to 1298 psig on recirculation flow, and that the turbine-driven pump develops a discharge pressure of greater than or equal to 1342 psig on recirculation flow when the secondary steam supply pressure is greater than 880 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump.

2. 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.

SURVEILLANCE REQUIREMENTS (Continued)

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 emergency feedwater actuation test signal.

   2. Verifying that each pump starts automatically upon receipt of an emergency feedwater actuation test signal.

c. Following any cold shutdown of 30 days or longer or whenever feedwater line cleaning through the emergency feedwater line has been performed, by verifying, by means of a flow test, the normal flow path from the condensate storage pool through each emergency feedwater pump to each of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven pump.

PLANT SYSTEMS

CONDENSATE STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage pool (CSP) shall be OPERABLE with a contained volume of at least 82% indicated level (170,000 gallons).

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the condensate storage pool inoperable, within 4 hours either:

a. Restore the CSP to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the wet cooling tower basins as a backup supply to the emergency feedwater pumps and restore the condensate storage pool to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage pool shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the pool is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The wet cooling tower basins shall be demonstrated OPERABLE at least once per 12 hours whenever the wet cooling tower basins are the supply source for the emergency feedwater pumps by verifying:

a. That each automatic and/or non-automatic valve in the flow path from the wet cooling tower basins to the emergency feedwater pumps is open or OPERABLE.

b. That the wet cooling tower basins contain a minimum contained water volume of 170,000 gallons.

c. That both auxiliary component cooling trains required by Specification 3.7.3 are OPERABLE and in operation.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcuries/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit.
	b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least HOT STANDBY within the next 6 hours.

MODES 2, 3, and 4

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 3.0 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 115°F when the pressure of the secondary coolant is greater than 210 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to less than or equal to 210 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 115°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure of the steam generators shall be determined to be less than 210 psig at least once per hour when the temperature of the secondary coolant is less than 115°F.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water and associated auxiliary component cooling water trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water and associated auxiliary component cooling water train OPERABLE, restore at least two trains 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 Each component cooling water and associated auxiliary component cooling water train 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 verifying that each automatic valve servicing safety-related equipment actuates to its correct position on SIAS and CSAS test signals.

c. At least once per 18 months by verifying that each component cooling water and associated auxiliary component cooling water pump starts automatically on an SIAS test signal.

PLANT SYSTEMS

3/4.7.4 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.4 Two independent trains of ultimate heat sink cooling towers shall be OPERABLE with each train consisting of a dry cooling tower (DCT) and a wet mechanical draft cooling tower and its associated water basin with:

a. A minimum water level in each wet tower basin of 97% (-9.86 ft MSL)

b. An average basin water temperature of less than or equal to 95°F.

c. Fans as required by Table 3.7-3.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one wet mechanical draft cooling tower inoperable due to low water level and/or high average water temperature, restore the wet mechanical draft cooling tower to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both wet mechanical draft cooling towers inoperable due to low water level and/or high average water temperature, restore at least one wet mechanical draft cooling tower to OPERABLE status within 1 hour and restore both wet mechanical draft cooling towers to OPERABLE status within 72 hours of initial loss, otherwise be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

c. With the number of OPERABLE fans less than required by Table 3.7-3, restore the number of OPERABLE fans to within the requirements of Table 3.7-3 within 72 hours (except as specified in Action e.), or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With the number of OPERABLE fans less than required by Table 3.7-3 on both DCT/WCT trains, restore the number of OPERABLE fans to within the requirements of Table 3.7-3 for 1 DCT/WCT train within 1 hour and comply with ACTION c. (except as specified in ACTION e.), or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

e. With one or more DCT fan(s) within the missile protected area of a DCT inoperable and if a Tornado Watch is in effect, restore the inoperable fan(s) to OPERABLE status within 1 hour, or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. With more than one DCT or WCT fan inoperable and the outside air temperature greater than 70°F, determine the dry bulb temperature at least once per hour. If the temperature exceeds either the dry or wet bulb limit of Table 3.7-3 for the number of fans that are inoperable, determine the wet bulb temperature and verify that the minimum fan requirements of Table 3.7-3 are satisfied.

SURVEILLANCE REQUIREMENTS

4.7.4. Each train of ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the average water temperature and water level to be within their limits.

b. At least once per 7 days, by verifying that each wet tower and dry tower fan that is not already running, starts and operates for at least 15 minutes.

TABLE 3.7-3

ULTIMATE HEAT SINK MINIMUM FAN REQUIREMENTS

AMBIENT CONDITION	DRY BULB $\geq$ 90°F OR WET BULB $\geq$ 81°F	80°F $\leq$ DRY BULB < 90°F AND WET BULB < 81°F	DRY BULB < 80°F AND WET BULB < 76°F
Fan Requirements(1)	14 DCT and 8 WCT OR 15 DCT and 7 WCT*	12 DCT*** and 4 WCT*	9 DCT** and 4 WCT*

(1) All fans of a dry cooling tower section are inoperable if component cooling water is secured to that section.

* Covers in place on out-of-service fans.

** With a tornado watch in effect, at least 8 of these 9 DCT fans under the missile protected portion of the DCT shall be OPERABLE.

*** With a tornado watch in effect, all DCT fans under the missile protected portion of the DCT shall be OPERABLE.

3/4.7.5 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.5 Flood protection shall be provided for all safety-related systems, components, and structures when the water level of the Mississippi River exceeds +27.0 ft Mean Sea Level USGS datum, at the levee fronting the Waterford Unit 3 site.

APPLICABILITY: At all times.

ACTION:

With the water level at the levee fronting the Waterford Unit 3 site above elevation +27.0 ft Mean Sea Level USGS datum initiate and complete within 12 hours procedures ensuring that all doors and penetrations below the +30.0 ft elevation are secure.

SURVEILLANCE REQUIREMENTS

4.7.5 The water level at the levee fronting the Waterford Unit 3 site shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is equal to or above elevation +24.0 ft Mean Sea Level USGS datum and below elevation +27.0 ft Mean Sea Level USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation +27.0 ft Mean Sea Level USGS datum.

3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent control room air conditioning systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3, and 4:

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

b. With both control room air conditioning systems inoperable, or with the OPERABLE control room air conditioning system, required to be in the recirculation mode by ACTION a, not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.6 Each control room air conditioning system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is maintained less than or equal to 110°F, by the operating system(s).

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 10 hours continuous with the heaters on.

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:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the air conditioning 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, and the system flow rate is 4225 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 4225 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by 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.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.8 inches water gauge while operating the system at a flow rate of 4225 cfm ± 10%.

2. Verifying that on a safety injection actuation test signal or a high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch water gauge relative to the outside atmosphere during system operation.

4. Verifying that the heaters dissipate 10 ± 1.0 kW when tested in accordance with ANSI N510-1975.

5. Verifying that on a toxic gas detection signal, the system automatically switches to the isolation mode of operation.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4225 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4225 cfm ± 10%.

3/4.7.7 CONTROLLED VENTILATION AREA SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent controlled ventilation area systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one controlled ventilation area 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.

SURVEILLANCE REQUIREMENTS

4.7.7 Each controlled ventilation area system 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 filters and charcoal adsorbers and verifying that the system operates for at least 10 hours continuous 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 controlled ventilation area 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, and the system flow rate is 3000 cfm $\pm$ 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 3000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position

SURVEILLANCE REQUIREMENTS (Continued)

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.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.8 inches water gauge while operating the system at a flow rate of 3000 cfm $\pm$ 10%.

2. Verifying that the system starts on a Safety Injection Actuation Test Signal and achieves and maintains a negative pressure of $\geq$ 0.25 inch water gauge within 45 seconds.

3. Verifying that the filter cooling bypass valves can be manually cycled.

4. Verifying that the heaters dissipate 20 ± 1.0 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 3000 cfm $\pm$ 10%.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those OPERATIONAL MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVELLIANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types
   As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections
   Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type on Any System 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 Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically suspectible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.8f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.8f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.8f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. 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;

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

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

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

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.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

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

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced 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 result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

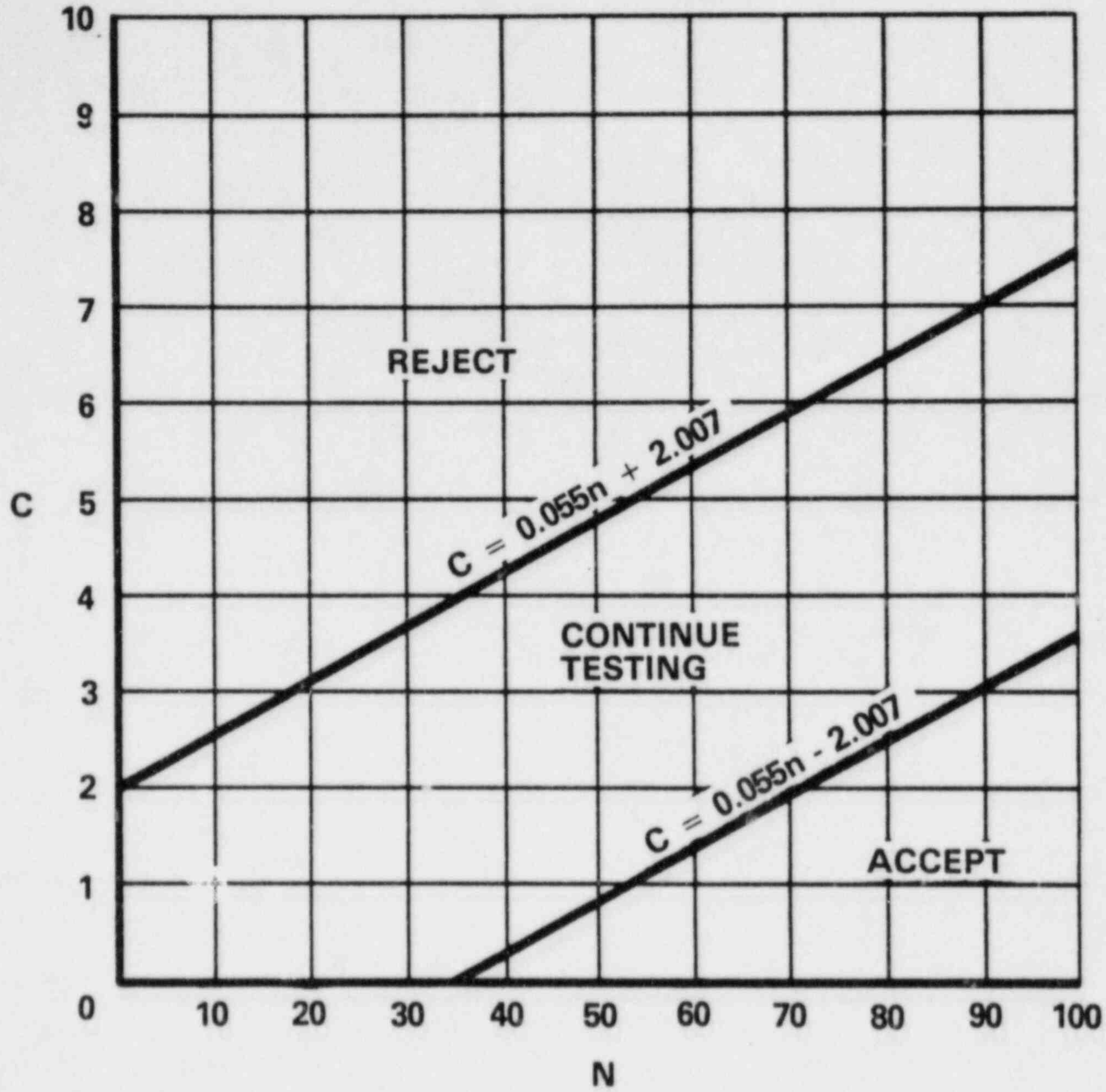

FIGURE 4.7-1

SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 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 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

    1. 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

4.7.9.1 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 microcurie per test sample.

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

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive material:

    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 6 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 installation and following repair or maintenance to the source or detector.

4.7.9.3 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 microcurie of removable contamination.

PLANT SYSTEMS

3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The fire suppression water system shall be OPERABLE with:

a. Two fire suppression pumps, each with a capacity of 2000 gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies, each with a minimum contained volume of 237,000 gallons (33 feet), and

c. An OPERABLE flow path capable of taking suction from the east fire water tank and the west fire water tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.10.2, 3.7.10.4, and 3.7.10.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore at least two pumps and/or water supplies to OPERABLE status within 7 days or provide an alternate backup pump or supply.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes.

c. 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.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 12 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each pump develops at least 2000 gpm at a total head of 100 psid by verifying at least 3 points on the pump performance curve during performance testing.

2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 96.5 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The diesel fuel oil day storage tank contains at least 170 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water and sediment.

c. At least once per 18 months during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.10.1.3 Each fire pump diesel starting 12-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each battery is above the plates, and

   2. The overall battery voltage is greater than or equal to 12 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

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

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

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

LIMITING CONDITION FOR OPERATION

3.7.10.2 The following spray and/or sprinkler systems shall be OPERABLE:

Sprinkler No.	Bldg./Elev.	Location
FPM-1	RCB	Reactor Coolant Pumps 1A, 1B
FPM-2	RCB	Reactor Coolant Pump 2A, 2B
FPM-3A	RAB +21, +46	Diesel Generator Area A, Feed Tank Room A
FPM-4B	RAB +21, +46	Diesel Generator Area B, Feed Tank Room B
FPM-11A	RAB -35	Emergency D/G Fuel Oil Tank A
FPM-12B	RAB -35	Emergency D/G Fuel Oil Tank B
FPM-16	FWPH +15	Fire Water Pump House
FPM-17	RAB +35	Cable Vault Area
FPM-18	RAB +35	Electrical Penetration Area 1
FPM-19	RAB +35	Electrical Penetration Area 2
FPM-22	RAB -4	Corridor and Blowdown Tank Rooms
FPM-23	RAB -35	Corridor, Shutdown Heat Exchanger Rooms, EFW Pump Room
FPM-24	RAB +21	Corridors, CCW Area
FPM-25B	RAB +21	North High Voltage Switchgear Room
FPM-26	RAB +46	Ventilation Equipment Rooms
FPM-27	RAB +7	HVAC Rooms
FPM-28	RAB -35	Auxiliary Component Cooling Water Pump Rooms
FPM-29	RAB +35	Relay Room, Corridor
FPM-30A	RAB +21	South High Voltage Switchgear Room

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged unless the spray and/or sprinkler system(s) is located inside the containment, then inspect that containment area at least once per 8 hours or monitor air temperature at least once per hour at the locations listed in Specification 4.6.1.5; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required spray and/or sprinkler systems 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. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a thermal/preaction test signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The computer room Halon system shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With the above required Halon system inoperable, within 1 hour establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.10.3 The above required Halon 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 is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight (level) and pressure to be at least 90% of full charge pressure.

c. At least once per 18 months by verifying:

   1. The system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated test signal, and

   2. Performance of a flow test through headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.4 The fire hose stations shown in Table 3.7-4 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 3.7-4 inoperable, provide gated wye(s) on the nearest operable hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the operable hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above action shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.10.4 Each of the fire hose stations shown in Table 3.7-4 shall be demonstrated OPERABLE:

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

b. At least once per 18 months by:

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

   2. Removing the hose for inspection and reracking, and

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

SURVEILLANCE REQUIREMENTS (Continued)

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 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION BLDG/COLUMN		ELEVATION (Feet MSL)	HOSE RACK IDENTIFICATION
FHB	2FH-U	-35	FH/A-010
FHB	3FH-V	+1	FH/A-020
FHB	5FH-V	+1	FH/A-021
FHB	2FH-T	+21	FH/A-040
FHB	2FH-V	+18	FH/A-041
FHB	Escape Exit	+21	FH/A-043
FHB	6FH-W	+18	FH/A-042
FHB	2FH-T	+46	FH/A-030
FHB	7FH-W	+46	FH/A-031
RAB	J-9A	-35	RA/F-101
RAB	J-6A	-35	RA/F-102
RAB	H-4A	-35	RA/F-103
RAB	J-3A	-35	RA/F-104
RAB	K-4A	-35	RA/F-105
RAB	M-10AZ	-35	RA/C-106
RAB	M-3A	-35	RA/A-107
RAB	M-2AC	-20	RA/K-108
RAB	M-11AZ	-20	RA/L-109
RAB	K-10A	-35	RA/F-110
RAB	K-11A	-4	RA/D-201
RAB	H-11A	-4	RA/D-202
RAB	J-10A	-4	RA/E-203
RAB	J-6A	-4	RA/H-204
RAB	H-4A	-4	RA/I-205
RAB	K-4A	-4	RA/J-206
RAB	M-10AZ	-4	RA/C-207
RAB	M-3A	-4	RA/A-208
RAB	LY-8A	-4	RA/B-209
RAB	K12A	+7	RA/D-301
RAB	J11A	+7	RA/D-302
RAB	H-11A	+21	RA/E-401
RAB	H-9A	+21	RA/E-402
RAB	K-11A	+21	RA/D-403
RAB	L-7A	+21	RA/C-411
RAB	N-10AZ	+21	RA/C-405
RAB	J-6A	+21	RA/I-406
RAB	H-4A	+21	RA/I-407
RAB	J-1A	+21	RA/I-408
RAB	K-4A	+21	RA/J-409
RAB	L-7A	+21	RA/G-410
RAB	N-4A	+21	RA/A-412
RAB	L-2A	+21	RA/A-413
RAB	H-12A	+35	RA/E-501
RAB	K-11A	+35	RA/E-502
RAB	N-10AZ	+35	RA/C-503
RAB	Northeast Stairwell	+21	RA/C-404

TABLE 3.7-4 (Continued)

FIRE HOSE STATIONS

LOCATION BLDG/COLUMN	ELEVATION (Feet MSL)	HOSE RACK IDENTIFICATION
RAB H-10A	+35	RA/E-504
RAB K-9A	+35	RA/E-505
RAB L-9A	+35	RA/C-506
RAB LY-6A	+35	RA/A-507
RAB J-9A	+35	RA/E-508
RAB L-8A	+35	RA/E-509
RAB G-9A	+35	RA/E-510
RAB J-12A	+46	RA/E-601
RAB K-11A	+46	RA/G-602
RAB K-10A	+46	RA/G-603
RAB K-8A	+46	RA/G-604
RAB G-7A	+46	RA/G-605
RAB K-6A	+46	RA/J-606
RAB K-2A	+46	RA/J-607
RAB J-2A	+46	RA/J-608
RAB J-3A	+46	RA/J-609
RAB K-2A	+69	RA/J-701
*RCB 19	-4	R/A-201
*RCB 12	-4	R/A-202
*RCB 6	-4	R/B-203
*RCB 1	-4	R/B-204
*RCB 20	+21	R/A-401
*RCB 13	+21	R/A-402
*RCB 7	+21	R/B-403
*RCB 1	+21	R/B-404
*RCB 13	+35	R/A-502
*RCB 20	+46	R/A-601
*RCB 13	+46	R/A-602
*RCB 7	+46	R/B-603
*RCB 2	+46	R/B-604

*Indicates all hose stations not accessible during plant operations and shall be demonstrated operable at least once per 18 months.

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.10.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be OPERABLE.

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

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2-1/2-inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hose is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.10.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

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

   3. Performing a flow check of each hydrant to verify its OPERABILITY.

   4. Visually inspecting each yard fire hydrant and verifying that the hydrant barrel and the hydrant are not damaged.

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION	HYDRANT NUMBER
Northeast Side RAB	4
Northeast Corner FHB	6
Northwest Corner FHB	7
Northeast Corner Service Building	9

PLANT SYSTEMS

3/4.7.11 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.11 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/ and associated hardware.

c. Performing a visual inspection of at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release, and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. That each locked-closed fire door is closed at least once per 7 days.

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test of these mechanisms at least once per 18 months.

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.12 Two independent essential services chilled water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With only one essential services chilled water loop OPERABLE, restore 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

4.7.12.1 Each of the above required essential services chilled water loop 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 31 days by verifying that the water outlet temperature is $\leq$ 42°F at a flow rate of $\geq$ 500 gpm.

c. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a safety injection actuation test signal.

d. At least once per 18 months, by verifying that each essential services chilled water pump and compressor starts automatically on a safety injection actuation test signal.

4.7.12.2 The backup essential services chilled water pump and chiller shall be demonstrated OPERABLE in accordance with Specification 4.7.12.1 whenever it is functioning as part of one of the required essential services chilled water loops.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

   1. Diesel oil feed tanks containing a minimum volume of 337 gallons of fuel, and

   2. A separate diesel generator fuel oil storage tank containing a minimum volume of 38,760 gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators 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.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30-hours.

c. With one diesel generator inoperable, in addition to ACTION a. or b. above, verify that:

   (1) All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   (2) When in MODE 1, 2, or 3, the steam-driven emergency feed pump is OPERABLE.

   If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION (Continued)

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months by transferring manually and automatically unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the diesel oil feed tank,

2. Verifying the fuel level in the diesel generator fuel oil storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the diesel oil feed tank,

4. Verifying the diesel starts from ambient condition and accelerates to at least 600 rpm (60 ± 1.2 Hz) in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 420 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:

   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 4400 kW in less than or equal to 176 seconds,* and operates with a load greater than or equal to 4400 kW for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

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 diesel oil feed tanks.

c. At least once per 92 days and from new fuel oil prior to addition to the storage tanks, by obtaining a sample of fuel oil in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

1. As soon as sample is taken (or prior to adding new fuel to the storage tank) verify in accordance with the test specified in ASTM-D975-77 that the sample has:

   a) A water and sediment content of less or equal to 0.05 volume percent.

   b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

   c) A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to 0.80 but less than or equal to 0.99 or an API gravity @ 60°F of greater than or equal to 11 degrees but less than or equal to 47 degrees.

*The diesel generator start (10 sec) and subsequent loading (176 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts and loading for the purpose of this surveillance testing may be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS (Continued)

2. Within 1 week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70.

3. Within 2 weeks of obtaining the sample verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137 Position 2.a are met when tested in accordance with ASTM-D975-77.

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 498 kW (HPSI pump) while maintaining voltage at 4160 ± 420 volts and frequency at 60 +4.5, -1.2 Hz.

3. Verifying the generator capability to reject a load of 4400 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

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

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds after the auto-start signal, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 + 1.2, -0.3 Hz during this test.

5. Verifying that on an SIAS actuation test signal (without loss-of-offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss-of-offsite power in conjunction with an SIAS actuation test signal, and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds after the auto-start signal, energizes the auto-connected emergency loads through the load sequencer and operates for greater than or equal to 5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 + 1.2, -0.3 Hz during this test.

   c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4840 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4400 kW. The generator voltage and freqency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 + 1.2, -0.3 Hz during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.d.4b.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4400 kW.

9. 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 standby status.

10. 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 energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the diesel oil feed tank of each diesel via the installed cross connection lines.

12. Verifying that the automatic load sequence timer is OPERABLE with the time of each load block within ±10% of the sequenced load block time.

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

    a) turning gear engaged
    b) emergency stop
    c) loss of D.C. control power
    d) governor fuel oil linkage tripped

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 600 rpm (60 ± 1.2 Hz) in less than or equal to 10 seconds.

f. At least once per 10 years by:

1. Draining each diesel generator fuel oil storage tank, removing the accumulated sediment, and cleaning the tank using a sodium hypochlorite solution or equivalent, and

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% of the system design pressure.

g. By performing a visual inspection of the interior of the diesel generator fuel oil storage tanks each time the tank is drained and, if necessary, clean the tank with a sodium hypochlorite solution, or equivalent.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. 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 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS.*	TEST FREQUENCY
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last 100 valid tests". Entry into this test schedule shall be made at the 31 day test frequency.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. A diesel oil feed tank containing a minimum volume of 337 gallons of fuel,

   2. The diesel fuel oil storage tanks containing a minimum volume of 38,760 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2, (except for Surveillance Requirement 4.8.1.1.2a.5.) and 4.8.1.1.3.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery Bank No. 3A-S and one associated full capacity charger (3A1-S or 3A2-S).

b. 125-volt Battery Bank No. 3B-S and one associated full capacity charger (3B1-S or 3B2-S).

c. 125-volt battery Bank No. 3AB-S and one associated full capacity charger (3AB1-S or 3AB2-S).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 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 of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and at least one associated charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

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

2. The total battery terminal voltage is greater than or equal to 125 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

2. 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. The average electrolyte temperature of a random sample of at least ten of the 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 anticorrosion material,

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

4. The battery charger will supply at least 150 amperes for 3A1-S, 3A2-S, 3B1-S and 3B2-S and 200 amperes for 3AB1-S and 3AB2-S at greater than or equal to 132 volts for at least 8 hours.

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

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

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

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	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 Average of all connected cells > 1.200	Not more than 0.020 below the average of all connected cells Average of all connected cells $\geq$ 1.190(b)

(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 Category A and B 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 the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value, declare the battery inoperable.

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amps when on charge.

(c) Corrected for average electrolyte temperature.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank (3A-S or 3B-S) and one associated full capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible.

b. With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by perfoming Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following Engineered Safety Features (ESF) and Static Uninterruptible Power Supply (SUPS) busses shall be energized in the specified manner. The tie breakers from the Train AB Busses shall be connected to either Train A or Train B.

a. Train A A.C. ESF busses consisting of:

 1. 4160-volt ESF Bus #3A3-S
 2. 480-volt ESF Bus #3A31-S

b. Train B A.C. ESF busses consisting of:

 1. 4160-volt ESF Bus #3B3-S
 2. 480-volt ESF Bus #3B31-S

c. Train AB A.C. ESF busses consisting of:

 1. 4160-volt ESF Bus #3AB3-S
 2. 480-volt ESF Bus #3AB31-S

d. 120-volt A.C. SUPS Bus #3MA-S energized from its associated inverter connected to D.C. Bus #3A-DC-S*.

e. 120-volt A.C. SUPS Bus #3MB-S energized from its associated inverter connected to D.C. Bus #3B-DC-S*.

f. 120-volt A.C. SUPS Bus #3MC-S energized from its associated inverter connected to D.C. Bus #3A-DC-S*.

g. 120-volt A.C. SUPS Bus #3MD-S energized from its associated inverter connected to D.C. Bus #3B-DC-S*.

h. 120-volt A.C. SUPS Bus #3A-S energized from its associated inverter connected to D.C. Bus #3A-DC-S.

i. 120-volt A.C. SUPS Bus #3B-S energized from its associated inverter connected to D.C Bus #3B-DC-S.

j. 125-volt D.C. Bus #3A-DC-S connected to Battery Bank #3A-S.

k. 125-volt D.C. Bus #3B-DC-S connected to Battery Bank #3B-S.

l. 125-volt D.C. Bus #3AB-DC-S connected to Battery Bank #3AB-S.

APPLICABILITY: MODES 1, 2, 3, and 4.

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*Two inverters may be disconnected from their D.C. bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their SUPS busses are energized, and (2) the SUPS busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. With one of the required divisions of A.C. ESF busses not fully energized, reenergize the division 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 A.C. SUPS bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. SUPS bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) reenergize the A.C. SUPS bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not connected to its associated battery bank, reconnect the D.C. bus from its associated OPERABLE battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One division of A.C. ESF busses consisting of one 4160 volt and one 480-volt A.C. ESF bus (3A3-S and 3A31-S or 3B3-S and 3B31-S).

b. Two 120-volt A.C. SUPS busses energized from their associated inverters connected to their respective D.C. busses (3MA-S, 3MB-S, 3MC-S, or 3MD-S).

c. One 120-volt A.C. SUPS Bus (3A-S or 3B-S) energized from its associated inverter connected to its respective D.C. bus.

d. One 125-volt D.C. bus (3A-DC-S or 3B-DC-S) connected to its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one or more of the above required containment penetration conductor overcurrent devices shown in Table 3.8-1 inoperable:

1. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping, racking out, or removing the alternate device or racking out or removing the inoperable device within 72 hours, and

2. Declare the affected system or component inoperable, and

3. Verify at least once per 7 days thereafter the alternate device is tripped, racked out, or removed, or the device is racked out or removed.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices which have the inoperable device racked out or removed or, which have the alternate device tripped, racked out, or removed.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-1.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers, except as noted on Table 3.8-1, shall consist of injecting a current in excess of the breakers' nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-1

CONTAINMENT PENETRATION

CONDUCTOR OVERCURRENT

PROTECTIVE DEVICES

6.9 kV POWER FROM MEDIUM VOLTAGE SWITCHGEAR

ITEM NO.	SYSTEM POWERED	BREAKER PROTECTION	OVERCURRENT PROTECTIVE DEVICES (NOTE 3) DEVICE TYPE AND TRIP SETPOINT (NOTE 1) SHEET NO.	LINE NO.	TIME-CURRENT CHARACTERISTIC	REMARKS
1	Reactor Coolant Pump 1A	Primary	11A1	15, 16, 17	Note 2	Items 1 thru 4 - Backup protection is provided by Transfer Trip Relays to Startup Transformer and Unit Auxiliary Transformer Breakers.
		Backup 1 Backup 2	Adjust Transfer Trip Relay 2/220 to 4 s (TDPU)			
2	Reactor Coolant Pump 1B	Primary	12A1	15, 16, 17	Note 2	
		Backup 1 Backup 2	Adjust Transfer Trip Relay 2/230 to 4 s (TDPU)			
3	Reactor Coolant Pump 2A	Primary	11A1	18, 19, 20	Note 2	
		Backup 1 Backup 2	Adjust Transfer Trip Relay 2/240 to 4 s (TDPU)			
4	Reactor Coolant Pump 2B	Primary	12A1	18, 19, 20	Note 2	
		Backup 1 Backup 2	Adjust Transfer Trip Relay 2/250 to 4 s (TDPU)			
						Operation of Primary and Backup Overcurrent Protection is illustrated on FSAR Figure 8.3-28.

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM LOW VOLTAGE SWITCHGEAR

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES (NOTE 6) TYPE AND TRIP SETPOINT (NOTE 1) SHEET NO.	LINE NO.	TIME-CURRENT CHARACTERISTIC	REMARKS
1	Polar Crane	Primary Backup	See Remarks			Item 1 - Primary Breaker is Locked Out in the Open Position during MODES 1, 2, 3, and 4. Therefore, inoperable Primary or Backup Protection is not a LCO.
2	CEDM Cooling Unit E-16(3A)	Primary (Note 4) Backup	20A1,20A2 Adjust Transfer Trip Relay 2/1139 to 1 s.	17, 17	Notes 2, 3	Items 2 through 5 - The Backup Protection is accomplished by Transfer Trip Relays as illustrated on FSAR Figure 8.3-29.
3	CEDM Cooling Unit E-16(3C)	Primary (Note 4) Backup	20A3,20A4 Adjust Transfer Trip Relay 2/1140 to 1 s.	23, 23	Notes 2, 3	
4	CEDM Cooling Unit E-16(3D)	Primary (Note 4) Backup	21A3,21A4 Adjust Transfer Trip Relay 2/1142 to 1 s.	24, 24	Notes 2, 3	
5	CEDM Cooling Unit E-16(3B)	Primary (Note 4) Backup	21A1,21A2 Adjust Transfer Trip Relay 2/1141 to 1 s.	18, 18	Notes 2, 3	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM LOW VOLTAGE SWITCHGEAR (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES (NOTE 6) TYPE AND TRIP SETPOINT (NOTE 1) SHEET NO.	LINE NO.	TIME-CURRENT CHARACTERISTIC	REMARKS
6	Press. Heaters Backup Bank 1 (B-1)	Primary (Note 4)	23A1,23A2	4, 4	Notes 2, 5	Items 6 through 11 - The Backup Protection consists of Transfer Trip Relays activated by any one of the Primary Overcurrent Protective Relays as illustrated on FSAR Figure 8.3-30.
		Backup	Adjust Transfer Trip Relay 2/285 to 0.5 s.			
7	Press. Heaters Backup Bank 2 (B-2)	Primary (Note 4)	23A1,23A2	5, 5	Notes 2, 5	
		Backup	Adjust Transfer Trip Relay 2/286 to 0.5 s.			
8	Press. Heaters Backup Bank 3 (B-3)	Primary (Note 4)	23A1,23A2	6, 6	Notes 2, 5	
		Backup	Adjust Transfer Trip Relay 2/287 to 0.5 s.			
9	Press. Heaters Backup Bank 4 (B-4)	Primary (Note 4)	24A1,24A2	4, 4	Notes 2, 5	
		Backup	Adjust Transfer Trip Relay 2/288 to 0.5 s.			
10	Press. Heaters Backup Bank 5 (B-5)	Primary (Note 4)	24A1,24A2	5, 5	Notes 2, 5	
		Backup	Adjust Transfer Trip Relay 2/289 to 0.5 s.			
11	Press. Heaters Backup Bank 6 (B-6)	Primary (Note 4)	24A1,24A2	6, 6	Notes 2, 5	
		Backup	Adjust transfer Trip Relay 2/290 to 0.5 s.			

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM LOW VOLTAGE SWITCHGEAR (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES: TYPE AND TRIP SETPOINT (NOTE 1) SHEET NO.	LINE NO.	OVERCURRENT PROTECTIVE DEVICES: TIME-CURRENT CHARACTERISTIC	REMARKS
12	Press. Heaters Proportional Bank 1 (P-1)	Primary (Note 4) Backup	23A1,23A2 See Remarks	8, 8	Notes 2, 5	Items 12 and 13 - The Backup Protection is located in the Proportional Heater Local Control Panel and consists of CHASE-SHAWMUT Fuses A50P200.
13	Press. Heaters Proportional Bank 2 (P-2)	Primary (Note 4) Backup	24A1,24A2 See Remarks	8, 8	Notes 2, 5	

TABLE 3.8-1 (Continued)

240 VOLTS CEDM POWER

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TYPE AND TRIP SETPOINT (NOTES 3&4)* SHEET NO.	LINE NO.	TIME-CURRENT CHARACTERISTIC	REMARKS
1	CEDM Coils (91 Circuits)	Primary	Breaker Heineman 40A (30A for Hold Bus)	-	Heineman Series AM, Curve 3	240 V, 3Ø power is fed from the C-E Reactor Trip Switchgear to the CEDM Cabinets. The 91 circuits are separated into subgroups and hold busses. Each subgroup/hold bus is protected by 1 breaker and 3 fuses.
		Backup	Fuse 60A International Rectifier Cat. No. SF25 x 60	-	International Rectifier SF25 Series Curves	These cabinets feed power to the CEDM Coils via #4 AWG and #8 AWG Penetration Conductors.

*See notes from Table 3.8-1 for 480-volt power from MCCs.

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs

			OVERCURRENT PROTECTIVE DEVICES				
ITEM NO.	SYSTEM POWERED	PROTECTION	DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
1	Safety Inj. Tank 1A Iso. Val. 1SI-V1505 Tk 1A (SI-331A)	Primary	Breaker	EF	61	Notes 2, 3	
		Backup	Fuse	TRS	61	Note 4	
2	Safety Inj. Tank 2A Iso. Val. 1SI-V1507 Tk 2A (SI-332A)	Primary	Breaker	EF	61	Notes 2, 3	
		Backup	Fuse	TRS	61	Note 4	
3	LP-311	Primary	Breaker	EF	62	Notes 2, 3	
		Backup	Fuse	TRS	62	Note 4	
4	RCS Loop 2 SDC Iso. Val. 1SI-V1504A (SI-401A)	Primary	Breaker	EF	63	Notes 2, 3	
		Backup	Fuse	TRS	63	Note 4	
5	CARS Suction Val. 2HV-F253A (CARS-201A)	Primary	Breaker	EF	64	Notes 2, 3	
		Backup	Fuse	TRS	64	Note 4	
6	Hydraulic Pump For Val. 1SI-V1503A (SI-405A)	Primary	Breaker	EF	64	Notes 2, 3	
		Backup	Fuse	TRS	64	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
7	Safety Inj. Tank 1B Iso. Val. 1SI-V1506 Tk 1B (SI-331B)	Primary	Breaker	EF	65	Notes 2, 3	
		Backup	Fuse	TRS	65	Note 4	
8	Safety Inj. Tank 2B Iso. Val. 1SI-V1508 Tk 1B (SI-332B)	Primary	Breaker	EF	65	Notes 2, 3	
		Backup	Fuse	TRS	65	Note 4	
9	LP-310	Primary	Breaker	EF	66	Notes 2, 3	
		Backup	Fuse	TRS	66	Note 4	
10	RCS Loop 1 SDC Iso. Val. 1SI-V1502B (SI-401B)	Primary	Breaker	EF	67	Notes 2, 3	
		Backup	Fuse	TRS	67	Note 4	
11	CARS Suction Val. 2HV-F254B (CAR-201B)	Primary	Breaker	EF	68	Notes 2, 3	
		Backup	Fuse	TRS	68	Note 4	
12	Hydraulic Pump For Val. 1SI-V1501B (SI-405B)	Primary	Breaker	EF	68	Notes 2, 3	
		Backup	Fuse	TRS	68	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
13	Cont. 30KVA Transf.	Primary	Breaker	EF	71	Notes 2, 3	
	PDP 377A	Backup	Fuse	TRS	71	Note 4	
14	RCP 2A Oil Lift	Primary	Breaker	EF	71	Notes 2, 3	
	Pump A	Backup	Fuse	TRS	71	Note 4	
15	RCP 1A Oil Lift	Primary	Breaker	EF	71	Notes 2, 3	
	Pump A	Backup	Fuse	TRS	71	Note 4	
16	SG 1 Vent Val.	Primary	Breaker	EF	71	Notes 2, 3	
	2MS-V668 (MS-101A)	Backup	Fuse	TRS	71	Note 4	
17	Moveable Detector Drive	Primary	Breaker	EF	72	Notes 2, 3	
	Mach. 1	Backup	Fuse	TRS	72	Note 4	
18	SG 2 Vent. Val.	Primary	Breaker	EF	74	Notes 2, 3	
	2MS-V667 (MS-101B)	Backup	Fuse	TRS	74	Note 4	
19	RCP 1B Oil Lift	Primary	Breaker	EF	74	Notes 2, 3	
	Pump A	Backup	Fuse	TRS	74	Note 4	
20	RCP 2B Oil Lift	Primary	Breaker	EF	74	Notes 2, 3	
	Pump A	Backup	Fuse	TRS	74	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
21	Moveable Detector Drive Mach. 2	Primary	Breaker	EF	74	Notes 2, 3	
		Backup	Fuse	TRS	74	Note 4	
22	Cont. 30KVA Transf. PDP 378B	Primary	Breaker	EF	75	Notes 2, 3	
		Backup	Fuse	TRS	75	Note 4	
23	H_2 Recombiner Power Supply A	Primary	Breaker	FJ	77	Notes 2, 3	
		Backup	Fuse	TRS	77	Note 4	
24	Reactor Cavity Cooling Sys. Fan S-2 (3A)	Primary	Breaker	EF	78	Notes 2, 3	
		Backup	Fuse	TRS	78	Note 4	
25	Radiation Removal Unit E-13 (3A)	Primary	Breaker	EF	78	Notes 2, 3	
		Backup	Fuse	TRS	78	Note 4	
26	RCP 1A Oil Lift Pump B	Primary	Breaker	EF	78	Notes 2, 3	
		Backup	Fuse	TRS	78	Note 4	
27	RCP 2A Oil Lift Pump B	Primary	Breaker	EF	78	Notes 2, 3	
		Backup	Fuse	TRS	78	Note 4	
28	H_2 Recombiner Power Supply B	Primary	Breaker	FJ	80	Notes 2, 3	
		Backup	Fuse	TRS	80	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
29	Reactor Cavity Cooling Sys. Fan S-2 (3B)	Primary	Breaker	EF	81	Notes 2, 3	
		Backup	Fuse	TRS	81	Note 4	
30	Radiation Removal Unit E-13 (3B)	Primary	Breaker	EF	81	Notes 2, 3	
		Backup	Fuse	TRS	81	Note 4	
31	RCP 1B Oil Lift Pump B	Primary	Breaker	EF	81	Notes 2, 3	
		Backup	Fuse	TRS	81	Note 4	
32	RCP 2B Oil Lift Pump B	Primary	Breaker	EF	81	Notes 2, 3	
		Backup	Fuse	TRS	81	Note 4	
33	Missile Shield Truck Receptacle	Primary		See Remarks			Item 33 - Primary Breaker is Locked Out in the Open Position during MODES 1, 2, 3, and 4. Therefore, inoperable Primary or Backup Protection is not a LCO.
		Backup					
34	Cont. Cooling Unit AH-1(3A-SA)	Primary	Breaker	JL	97	Notes 2, 3	
		Backup (Note 5)	Breaker Relay	ECS IAC66T	20A1 20A2	Notes 6, 7, 8	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES			TIME-CURRENT CHARACTERISTIC	REMARKS
			DEVICE	TYPE	TRIP SETPOINT (NOTE 1)		
35	Cont. Cooling Unit AH-1 (3C-SA)	Primary	Breaker	JL	97	Notes 2, 3	
		Backup (Note 5)	Breaker Relay	ECS IAC66T	20A1 20A2	Notes 6, 7, 8	
36	Cont. Cooling Unit AH (3B-SB)	Primary	Breaker	JL	97	Notes 2, 3	
		Backup (Note 5)	Breaker Relay	ECS IAC66T	21A1 21A2	Notes 6, 7, 8	
37	Cont. Cooling Unit AH-1 (3D-SB)	Primary	Breaker	JL	97	Notes 2, 3	
		Backup (Note 5)	Breaker Relay	ECS IAC66T	21A1 21A2	Notes 6, 7, 8	
38	Cont. Sump Pump A	Primary	Breaker	EF	45	Notes 2, 3	
		Backup	Fuse	TRS	45	Note 4	
39	LP-306	Primary	Breaker	EF	45	Notes 2, 3	
		Backup	Fuse	TRS	45	Note 4	
40	LP-301	Primary	Breaker	EF	45	Notes 2, 3	
		Backup	Fuse	TRS	45	Note 4	
41	LP-302	Primary	Breaker	EF	45	Notes 2, 3	
		Backup	Fuse	TRS	45	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
42	LP-304	Primary	Breaker	EF	45	Notes 2, 3	
		Backup	Fuse	TRS	45	Note 4	
43	Cont. Elevator D	Primary	Breaker	EF	47	Notes 2, 3	
		Backup	Fuse	TRS	47	Note 4	
44	Refueling Cavity Drain Pump	Primary	Breaker	EF	48	Notes 2, 3	
		Backup	Fuse	TRS	48	Note 4	
45	Refueling Equipment	Primary	Breaker	EF	50	Notes 2, 3	
		Backup	Fuse	CRS	50	Note 4	
46	Refueling Equipment	Primary	Breaker	EF	48	Notes 2, 3	
		Backup	Fuse	TRS	48	Note 4	
47	Cont. Sump Pump B	Primary	Breaker	EF	49	Notes 2, 3	
		Backup	Fuse	TRS	49	Note 4	
48	LP-303	Primary	Breaker	EF	49	Notes 2, 3	
		Backup	Fuse	TRS	49	Note 4	
49	LP-305	Primary	Breaker	EF	49	Notes 2, 3	
		Backup	Fuse	TRS	49	Note 4	

TABLE 3.8-1 (Continued)

480 VOLTS POWER FROM MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES: DEVICE	TYPE	TRIP SETPOINT (NOTE 1)	TIME-CURRENT CHARACTERISTIC	REMARKS
50	LP-300	Primary	Breaker	EF	49	Notes 2, 3	
		Backup	Fuse	TRS	49	Note 4	
51	SDC Loop 1 Vacuum Priming Pump	Primary	Breaker	EF-3	45	Notes 2, 3	
		Backup	Fuse	TRS	43	Note 4	
52	SDC Loop 2 Vacuum Priming Pump	Primary	Breaker	EF-3	47	Notes 2, 3	
		Backup	Fuse	TRS	47	Note 4	
53	PDP 365A Receptacles	Primary	Breaker	TED	104	Notes 2, 3	
		Backup	Breaker	TED	104	Notes 2, 3	
54	PDP 366B Receptacles	Primary	Breaker	TED	104	Notes 2, 3	
		Backup	Breaker	TED	104	Notes 2, 3	

TABLE 3.8-1 (Continued)

208 VOLTS CONTROL POWER FROM PDPs OR MCCs

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES SHEET NO.	DEVICE	TYPE/CHARACTERISTIC (NOTE 2)
1	RCP 1A Heater	Primary	CWD 2269	Breaker	TEB
		Backup	CWD 2269	Breaker	TEB
2	RCP 2A Heater	Primary	CWD 2269	Breaker	TEB
		Backup	CWD 2269	Breaker	TEB
3	RCP 1B Heater	Primary	CWD 2270	Breaker	TEB
		Backup	CWD 2270	Breaker	TEB
4	RCP 2B Heater	Primary	CWD 2270	Breaker	TEB
		Backup	CWD 2270	Breaker	TEB

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES: TRIP SETPOINT (NOTE 1) SHEET NO.	OVERCURRENT PROTECTIVE DEVICES: TRIP SETPOINT (NOTE 1) CIRCUIT NO.	OVERCURRENT PROTECTIVE DEVICES: DEVICE	OVERCURRENT PROTECTIVE DEVICES: TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
1	Sol. Valve 1SI-F1551TK1A	Primary	186	Ckt. 26	Breaker	CD	
	(SI-303A)	Backup	186A	Ckt. 26	Fuse	FRN	
2	Sol. Valve 1SI-F1553TK2A	Primary	186	Ckt. 38	Breaker	CD	
	(SI-304A)	Backup	186A	Ckt. 38	Fuse	FRN	
3	Sol. Valve 2CC-F243AB	Primary	108A	Ckt. 4	Fuse	FRN	Two fuses in series, one each, + and
	(CC-710)	Backup	108A	Ckt. 4	Fuse	FRN	- poles.
4	Sol. Valve 2SI-F1561AB	Primary	186	Ckt. 5	Breaker	CD	
	(SI-343)	Backup	186A	Ckt. 5	Fuse	FRN	
5	Sol. Valve 2SI-F605TK1A	Primary	186	Ckt. 16	Breaker	CD	
	(NG-161A)	Backup	186A	Ckt. 16	Fuse	FRN	
6	Sol. Valve 2SI-F607TK2A	Primary	186	Ckt. 25	Breaker	CD	
	(NG-162A)	Backup	186A	Ckt. 25	Fuse	FRN	
7	Sol. Valve 2SI-E636	Primary	186	Ckt. 30	Breaker	CD	
	(SI-323A)	Backup	186A	Ckt. 30	Fuse	FRN	
8	Sol. Valve 2SI-E638	Primary	186	Ckt. 36	Breaker	CD	
	(SI-324A)	Backup	186A	Ckt. 36	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
9	Sol. Valve 1CH-F1516AB (CVC-101)	Primary	147	Ckt. 1	Breaker	CD	
		Backup	147A	Ckt. 1	Fuse	FRN	
10	Sol. Valve 1SI-V2504 (SI-301)	Primary	147	Ckt. 30	Breaker	CD	
		Backup	147A	Ckt. 30	Fuse	FRN	
11	Sol. Valve 2SI-F1564TK1A (SI-307A)	Primary	186	Ckt. 28	Breaker	CD	
		Backup	186A	Ckt. 28	Fuse	FRN	
12	Cont. Purge Iso. Sol. Valves 2HV-B151A (CAP-103) and 2HV-B152A (CAP-104)	Primary	120	Ckt. 26	Breaker	EE	
		Backup	120A	F1	Fuse	TRS	
13	H_2 Analyzer Valves A Power	Primary	120	Ckt. 7	Breaker	EE	
		Backup	1564-2084	CB 2	Breaker	P-15	Backup in Hydrogen Analyzer Panel, Breaker CB 2.
14	Cont. Spray Riser Pump A Sol. Valve 2CS-E608A (CS-129A)	Primary	120	Ckt. 9	Breaker	EE	
		Backup	120A	F3	Fuse	TRS	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
15	Sol. Valve 2SI-F1566TK2A	Primary	186	Ckt. 40	Breaker	CD	
	(SI-308A)	Backup	186A	Ckt. 40	Fuse	FRN	
16	Sol. Valve 2SI-E633	Primary	186	Ckt. 34	Breaker	CD	
	(SI-323B)	Backup	186A	Ckt. 34	Fuse	FRN	
17	Sol. Valve 2SI-E635	Primary	186	Ckt. 27	Breaker	CD	
	(SI-326B)	Backup	186A	Ckt. 27	Fuse	FRN	
18	Sol. Valves for 1SI-1503A	Primary	108A	Ckt. 7	Fuse	FRN	Two fuses in series, one each, + and
	(SI-405A)	Backup	108A	Ckt. 7	Fuse	FRN	- poles.
19	Sol. Valve 2HV-B156A	Primary	147	Ckt. 14	Breaker	CD	
	(CVR-201)	Backup	147A	Ckt. 14	Fuse	FRN	
20	Cont. Fan Coolers	Primary	120	Ckt. 17	Breaker	EE	
	Dampers	Backup	120A	F6	Fuse	TRS	
21	Motor Htr. Leads	Primary	120	Ckt. 13	Breaker	EE	
	AH-1 (3A-SA)	Backup	120A	F4	Fuse	TRS	
22	Motor Htr. Leads	Primary	120	Ckt. 15	Breaker	EE	
	AH-1 (3C-SA)	Backup	120A	F5	Fuse	TRS	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
23	Motor Htr. Leads E-16 (3A)	Primary	CWD1139		Breaker	TED	120/208V SWGR heater bus, double breaker protection.
		Backup	CWD1139		Breaker	TED	
[illegible]	Motor Htr. Leads E-16 (3C)	Primary	CWD1140		Breaker	TED	120/208V SWGR heater bus, double breaker protection.
		Backup	CWD1140		Breaker	TED	
25	Sol. Valves 1SI-F1552TK1B (SI-303B)	Primary	187	Ckt. 26	Breaker	CD	
		Backup	187A	Ckt. 26	Fuse	FRN	
26	Sol. Valve 1SI-F1554TK2B (SI-304B)	Primary	187	Ckt. 38	Breaker	CD	
		Backup	187A	Ckt. 38	Fuse	FRN	
27	Sol. Valve 2WM-F157AB (GWM-104)	Primary	187	Ckt. 7	Breaker	CD	
		Backup	187A	Ckt. 7	Fuse	FRN	
28	Sol. Valve 2SI-F606TK1B (NG-161B)	Primary	187	Ckt. 16	Breaker	CD	
		Backup	187A	Ckt. 16	Fuse	FRN	
29	Sol. Valve 2SI-F608TK2B (NG-162B)	Primary	187	Ckt. 25	Breaker	CD	
		Backup	187A	Ckt. 25	Fuse	FRN	
30	Sol. Valve 2SI-E637 (SI-325B)	Primary	187	Ckt. 30	Breaker	CD	
		Backup	187A	Ckt. 30	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
31	Sol. Valve 2SI-E639 (SI-324B)	Primary	187	Ckt. 27	Breaker	CD	
		Backup	187A	Ckt. 27	Fuse	FRN	
32	Sol. Valve 2CH-F1513AB (RC-606)	Primary	187	Ckt. 2	Breaker	CD	
		Backup	187A	Ckt. 2	Fuse	FRN	
33	Sol. Valve 1CH-F2501AB (CVC-103)	Primary	148	Ckt. 1	Breaker	CD	
		Backup	148A	Ckt. 1	Fuse	FRN	
34	Sol. Valve 1SI-V2505 (SI-302)	Primary	148	Ckt. 28	Breaker	CD	
		Backup	148A	Ckt. 28	Fuse	FRN	
35	Sol. Valve 2BM-F108AB (BM-109)	Primary	187	Ckt. 1	Breaker	CD	
		Backup	187A	Ckt. 1	Fuse	FRN	
36	Cont. Purge Iso. Sol. Valves 2HV-B154B (CAP-204) and 2HV-B153B (CAP-203)	Primary	121	Ckt. 26	Breaker	EE	
		Backup	121A	F6	Fuse	TRS	
37	Sol. Valve 2HV-B157B (CVR-101)	Primary	148	Ckt. 14	Breaker	CD	
		Backup	148A	Ckt. 14	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
38	Sol. Valve 2BD-F603 (BD-102A)	Primary	187	Ckt. 6	Breaker	CD	
		Backup	187A	Ckt. 6	Fuse	FRN	
39	Sol. Valve 2BD-F605 (BD-102B)	Primary	187	Ckt. 8	Breaker	CD	
		Backup	187A	Ckt. 8	Fuse	FRN	
40	H_2 Analyzer Valves B Power	Primary	121	Ckt. 7	Breaker	EE	
		Backup	1564-2084	CB 2	Breaker	P-15	Backup in Hydrogen Analyzer Panel Breaker CB 2.
41	Cont. Spray Riser Pump B Sol. Valve 2CS-E609B (CS-129B)	Primary	121	Ckt. 9	Breaker	EE	
		Backup	121A	F2	Fuse	TRS	
42	Cont. Sump Iso. Valve 2WM-F104AB (SP-105)	Primary	187	Ckt. 9	Breaker	CD	
		Backup	187A	Ckt. 9	Fuse	FRN	
43	Sol. Valve 2SI-F1565TK1B (SI-307B)	Primary	187	Ckt. 28	Breaker	CD	
		Backup	187A	Ckt. 28	Fuse	FRN	
44	Sol. Valve 2SI-F1567TK2B (SI-308B)	Primary	187	Ckt. 40	Breaker	CD	
		Backup	187A	Ckt. 40	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES — TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
45	Sol. Valve 2SI-E632	Primary	187	Ckt. 36	Breaker	CD	
	(SI-325A)	Backup	187A	Ckt. 36	Fuse	FRN	
46	Sol. Valve 2SI-E634	Primary	187	Ckt. 34	Breaker	CD	
	(SI-326A)	Backup	187A	Ckt. 34	Fuse	FRN	
47	Samp Sys. Sol. Valve	Primary	187	Ckt. 29	Breaker	CD	
	2SL-F1501AB (PSL-105)	Backup	187A	Ckt. 29	Fuse	FRN	
48	Samp. Sys. Sol. Valve	Primary	187	Ckt. 31	Breaker	CD	
	2SL-F1502AB (PSL-203)	Backup	187A	Ckt. 31	Fuse	FRN	
49	Samp. Sys. Sol. Valve	Primary	187	Ckt. 33	Breaker	CD	
	2SL-F1503AB (PSL-303)	Backup	187A	Ckt. 33	Fuse	FRN	
50	Sol. Valves for 1SI-1501B	Primary	109A	Ckt. 9	Fuse	FRN	Two fuses in series, one each, + and - poles.
	(SI-405B)	Backup	109A	Ckt. 9	Fuse	FRN	
51	Motor Htr. Leads	Primary	121	Ckt. 13	Breaker	EE	
	AH-1 (3B-SB)	Backup	121A	F3	Fuse	TRS	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
52	Motor Htr. Leads AH-1 (3D-SB)	Primary	121	Ckt. 15	Breaker	EE	
		Backup	121A	F4	Fuse	TRS	
53	Motor Htr. Leads E-16 (3B)	Primary	CWD 1141		Breaker	TED	120/208V SWGR heater bus, double breaker protection.
		Backup	CWD 1141		Breaker	TED	
54	Motor Htr. Leads E-16 (3D)	Primary	CWD 1142		Breaker	TED	120/208V SWGR heater bus, double breaker protection.
		Backup	CWD 1142		Breaker	TED	
55	Cont. Fan Coolers Dampers	Primary	121	Ckt. 17	Breaker	EE	
		Backup	121A	F5	Fuse	TRS	
56	Samp. Sys. Sol. Valve 2SL-F601 (PSL-404A)	Primary	148A	Ckt. 49	Breaker	CD	
		Backup	148A	Ckt. 49	Fuse	FRN	
57	Samp. Sys. Sol. Valve 2SL-F603 (PSL-404B)	Primary	148	Ckt. 45	Breaker	CD	
		Backup	148A	Ckt. 45	Fuse	FRN	
58	Samp. Sys. Recorder Panel	Primary	133	Ckt. 35	Breaker	EE	
		Backup	133A	F12	Fuse	TRS	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
59	Cont. Purge Exh. Damper	Primary	133	Ckt. 1	Breaker	EE	
	SV-D22 (CAP-202) and	Backup	133A	F5	Fuse	TRS	
	SV-D23 (CAP-201)	Primary	134	Ckt. 1	Breaker	EE	
		Backup	134A	F2	Fuse	ATM	
60	Sol. Valve 7RC-F604	Primary	133	Ckt. 8	Breaker	EE	
	(RC-323)	Backup	133A	F3	Fuse	TRS	
61	Sol. Valve 7RC-F605	Primary	133	Ckt. 10	Breaker	EE	
	(RC-325)	Backup	133A	F4	Fuse	TRS	
62	Sol. Valve 1CH-E2504B	Primary	148	Ckt. 29	Breaker	CD	
	(CVC-218B)	Backup	148A	Ckt. 29	Fuse	FRN	
63	Sol. Valve 1CH-E2503A	Primary	147	Ckt. 27	Breaker	CD	
	(CVC-218A)	Backup	147A	Ckt. 27	Fuse	FRN	
64	Sol. Valves 3CC-P1501A1	Primary	150	Ckt. 25	Breaker	TEB	
	(CC-665A) & 3CC P1505A1 (CC-679A)	Backup	CWD 280	F1	Fuse	ATM	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	 CIRCUIT NO.	 DEVICE	 TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
65	Sol. Valves 3CC-P1503A2 (CC-666A) & 3CC-P1507A2 (CC-680A)	Primary	150	Ckt. 27	Breaker	TEB	
		Backup	CWD 282	F1	Fuse	ATM	
66	RCP1A Instrumentation and Accessories	Primary	CWD 220		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 220		Fuse	OTS	
67	RCP2A Instrumentation and Accessories	Primary	CWD 240		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 240		Fuse	OTS	
68	CEDM Cool. Valves & Dampers	Primary	149	Ckt. 14	Breaker	TEB	
		Backup	CWD 1145	F2	Fuse	ATM	
69	CEDM Cool. Units Inlet Damper	Primary	150	Ckt. 20	Breaker	TEB	
		Backup	CWD 1145	F1	Fuse	ATM	
70	Sol. Valve 2CH-F1514AB (RC-602)	Primary	150	Ckt. 5	Breaker	TEB	
		Backup	CWD 326	F2	Fuse	ATM	
71	Sol. Valve 2BM-P237 (GWM-101)	Primary	135	Ckt. 11	Breaker	EE	
		Backup	CWD 401	F1	Fuse	[illegible]	

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

			OVERCURRENT PROTECTIVE DEVICES				
ITEM NO.	SYSTEM POWERED	PROTECTION	TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
72	Sol. Valve 5SI-F1563 (SI-342)	Primary	150	Ckt. 1	Breaker	TEB	
		Backup	CWD 499	F3	Fuse	ATM	
73	Sol. Valves 3CC-P1502B1 (CC-665B) & 3CC-P1506B1 (CC-679B)	Primary	150	Ckt. 26	Breaker	TEB	
		Backup	CWD 281	F2	Fuse	ATM	
74	Sol. Valves 3CC-P1504B2 (CC-666B) & 3CC-P1508B2 (CC-680B)	Primary	150	Ckt. 28	Breaker	TEB	
		Backup	CWD 283	F2	Fuse	ATM	
75	RCP1B Instrumentation and Accessories	Primary	CWD 230		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 230		Fuse	OTS	
76	RCP2B Instrumentation and Accessories	Primary	CWD 250		Fuse	OTS	Two fuses in series, one each, + and - poles.
		Backup	CWD 250		Fuse	OTS	
77	Sol. Valve 2CA-E604B (ARM-109)	Primary	148	Ckt. 26	Breaker	CD	
		Backup	148A	Ckt. 26	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
78	Sol. Valve 1CH-E2505A (CVC-216A)	Primary	147	Ckt. 31	Breaker	CD	
		Backup	147A	Ckt. 31	Fuse	FRN	
79	Sol. Valve 1CH-E2505B (CVC-216B)	Primary	148	Ckt. 31	Breaker	CD	
		Backup	148A	Ckt. 31	Fuse	FRN	
80	Sol. Valve 7WM-E677 (SP-102B)	Primary		CB 2	Breaker	CH	15a Breakers on Skid #4 (5817-6368)
		Backup		Ckt. H4	Breaker	QO	
81	Sol. Valves 2RC-2557A (RC-3184) 2RC-2559A (RC-1015) 2RC-2561A (RC-3186)	Primary	212	Ckt. 2	Breaker	EE	
		Backup	120A	F2	Fuse	TRS	
82	Sol. Valves 2RC-2558B (RC-3183) 2RC-2560B (RC-1014) 2RC-2562B (RC-1017)	Primary	213	Ckt. 2	Breaker	EE	
		Backup	121A	F1	Fuse	TRS	
83	1SI-V1505TK1A (SI-331A) Space HTR	Primary	186	Ckt. 13	Breaker	CD	
		Backup	186A	Ckt. 13	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
84	1SI-V1505TK1A (SI-331A)	Primary	147	Ckt. 6	Breaker	CD	
	Limit Switch & Ind. Lights	Backup	147A	Ckt. 6	Fuse	FRN	
85	1SI-V1507TK2A (SI-332A)	Primary	186	Ckt. 15	Breaker	CD	
	Space HTR	Backup	186A	Ckt. 15	Fuse	FRN	
86	1SI-V1507TK2A (SI-332A) Limit Switch	Primary	147	Ckt. 8	Breaker	CD	
	& Ind. Lights	Backup	147A	Ckt. 8	Fuse	FRN	
87	RCP 1A Speed Sensor	Primary	126	Ckt. 5	Breaker	EE	
		Backup	126A	F6	Fuse	ATM	
88	RCP 2A Speed Sensor	Primary	126	Ckt. 7	Breaker	EE	
		Backup	126A	F5	Fuse	ATM	
89	Radiation Removal Unit	Primary	133	Ckt. 24	Breaker	EE	
	E-13(3A) Thermistor	Backup	133A	F6	Fuse	TRS	
90	Cont. Cooling Unit	Primary	149	Ckt. 3	Breaker	TEB	
	Condensing Pot Flow Det.	Backup	CWD 829	F1	Fuse	ATM	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
91	Pzr Spray Valves 1RC-F1501A (RC301A)	Primary	150	Ckt. 4	Breaker	TEB	
	& 1RC-F1502B (RC301B)	Backup	CWD 296	F1	Fuse	ATM	
92	Movable Incore Det.	Primary	126	Ckt. 32	Breaker	EE	
	Drive Mach. #1 Control	Backup	CWD 158		Fuse	FRN	
93	Movable Incore Det.	Primary	136	Ckt. 7	Breaker	CD	
	Switching Device	Backup	CWD 158		Fuse	ABU	
94	Refueling Machine	Primary	5817-4241		Fuse	TRS	
	Control	Backup	5817-4241		Fuse	KTN/KTNR	
95	1SI-V1506TK1B (SI-331B)	Primary	187	Ckt. 13	Breaker	CD	
	Space HTR	Backup	187A	Ckt. 13	Fuse	FRN	
96	1SI-V1506TK1B (SI-331B)	Primary	148	Ckt. 6	Breaker	CD	
	Limit Switch & Ind. Lights	Backup	148A	Ckt. 6	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
97	1SI-V1508TK2B (SI-332B)	Primary	187	Ckt. 15	Breaker	CD	
	Space HTR	Backup	187A	Ckt. 15	Fuse	FRN	
98	1SI-V1508TK2B (SI-332B)	Primary	148	Ckt. 8	Breaker	CD	
	Limit Switch & Ind. Lights	Backup	148A	Ckt. 8	Fuse	FRN	
99	RCP 1B Speed	Primary	127	Ckt. 5	Breaker	EE	
	Sensor	Backup	127A	F6	Fuse	ATM	
100	RCP 2B Speed	Primary	127	Ckt. 7	Breaker	EE	
	Sensor	Backup	127A	F5	Fuse	ATM	
101	Radiation Removal Unit	Primary	134	Ckt. 24	Breaker	EE	
	E-13 (3B) Thermistor	Backup	134A	F1	Fuse	ATM	
102	Cont. Air Locks Door	Primary	147	Ckt. 33	Breaker	CD	
	Pos. Ind.	Backup	147A	Ckt. 33	Fuse	FRN	
103	Cont. Air Locks Door	Primary	148	Ckt. 33	Breaker	CD	
	Pos. Ind.	Backup	148A	Ckt. 33	Fuse	FRN	

TABLE 3.8-1 (Continued)

120 VOLTS CONTROL POWER FROM PDPs OR MCCs (Continued)

ITEM NO.	SYSTEM POWERED	PROTECTION	OVERCURRENT PROTECTIVE DEVICES TRIP SETPOINT (NOTE 1) SHEET NO.	CIRCUIT NO.	DEVICE	TYPE/TIME-CURRENT CHARACTERISTIC (NOTE 2)	REMARKS
104	2BM-F108AB (BM-109)	Primary	133	Ckt. 34	Breaker	EE	
	Pos. Ind.	Backup	CWD 400/ 405	Ckt. 2	Fuse	NON	
105	2WM-F157AB (GWM-104)	Primary	133	Ckt. 33	Breaker	EE	
	Pos Ind.	Backup	CWD 650/ 680	Ckt. 19	Fuse	NON	
106	Movable Incore Det.	Primary	127	Ckt. 32	Breaker	EE	
	Drive Mach. #2 Control	Backup	CWD 158		Fuse	FRN	
107	CEDM Cooling Units	Primary	110A/CWD 1028	Ckt. 24	Fuse	FB	Two fused breakers, one each, + and - poles.
	Vibration Switches	Backup	110A/CWD 1028	Ckt. 24	Fuse	FB	

TABLE 3.8-1 (Continued)

NOTES

6.9 kV POWER FROM MEDIUM VOLTAGE SWITCHGEAR

1) Refer to drawing LOU-1564-B-289 sheet and line numbers as indicated.

2) Refer to G.E. curve in GEI-68751A and GEI-19959 instruction books for IAC 66M3A and IAC57 relays.

3) Relay testings to be performed in accordance with vendor's relay calibration procedures.

TABLE 3.8-1 (Continued)

NOTES (Continued)

480 VOLTS POWER FROM LOW VOLTAGE SWITCHGEAR

1) Refer to drawing LOU-1564-B-289 sheet and line numbers as indicated.

2) Refer to G.E. curve GES-6032A for ECS programmer.

3) Refer to G.E. curve in GEI-19959 instruction book for IAC57 relays.

4) Primary breaker is equipped with two sets of protective devices.

5) Refer to G.E. curve GES-7005A for IAC77 relays.

6) Relay and programmer testing to be performed in accordance with vendor's calibration procedures.

TABLE 3.8-1 (Continued)

NOTES (Continued)

480 VOLTS POWER FROM MCCs

1) Refer to drawing LOU-1564-B-289 sheet numbers as indicated. Circuit breakers with adjustable instantaneous magnetic trip element are set on the basis of two times the motor locked rotor current. For static loads the setpoint is the minimum available.

2) Refer to the appropriate curves as follows:

   EF, EH - ITE/Gould TD8087
   EF3 - ITE/Gould Instantaneous Trip
   FJ Breaker - ITE/Gould TD4948
   JL Breaker - ITE/Gould TD4950
   TED Breaker - GE GES-6114A

3) Circuit breaker testing to be performed in accordance with vendor's molded case breaker calibration procedures.

4) Fuse testing to be performed in accordance with vendor's nondestructive resistance test procedures.

5) Backup breaker is equipped with two sets of protective devices.

6) Refer to G.E. curve GES-6032A for ECS programmer.

7) Refer to G.E. curve GES-7004A for IAC66T relays.

8) Relay and programmer testing to be performed in accordance with vendor's calibration procedures.

9) Equivalent breakers and fuses may be substituted for the types specified.

TABLE 3.8-1 (Continued)

NOTES (Continued)

208 VOLTS AND 120 VOLTS CONTROL POWER FROM PDPs or MCCs

1) For trip setpoint, refer to drawing LOU-1564-B-289 sheet numbers as indicated.

2) Below is listing of molded case breakers by type giving the curve number for time-current characteristic:

TYPE	MANUFACTURER	CURVE NO.
EE, EF	ITE	TD 4947
CD	Heineman	CD, CE, CF
TEB	GE	GES-6122B, 6122
TED	GE	GES-6119C
AM	Heineman	AM
QO	Square D	630-2
CH	Cutler Hammer	Safety Breaker Curve

3) Equivalent breakers and fuses may be substituted for the types specified.

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection and bypass devices, integral with the motor starter, of each valve listed in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE.

a. At least once per 18 months, by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

    1. Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

    2. Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

    1. All thermal overload devices which are not bypassed, such that each nonbypassed device is calibrated at least once per 6 years.

    2. All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE and not bypassed, at least once per 6 years.

TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
2SI-V1541A2 (SI-139A)	LPSI Flow Control	Yes
2SI-V1543B2 (SI-138A)	LPSI Flow Control	Yes
2SI-V1550A1 (SI-225A)	HPSI Flow Control	Yes
2SI-V1542A3 (SI-227A)	HPSI Flow Control	Yes
3CH-V112A/B (BAM-133)	Reactor Makeup Bypass	Yes
2SI-V1546A2 (SI-226A)	HPSI Flow Control	Yes
2SI-V1548A4 (SI-228A)	HPSI Flow Control	Yes
1SI-V1504A (SI-401A)	RCS Loop 2 Shutdown Cooling Isolation	No
1SI-V1505TK1A (SI-331A)	Safety Inj. Tank 1A Isolation	Yes
1SI-V1507TK2A (SI-332A)	Safety Inj. Tank 2A Isolation	Yes
2HV-B158A (SBV-110A)	SBVS A Train Outlet	Yes
2HV-B160A (SBV-101A)	SBVS A Train Inlet	Yes
2HV-B162A (SBV-114A)	SBVS A Exhaust	Yes
2HV-B164A (SBV-113A)	SBVS A Recirc.	Yes
3HV-B196A (HVC-201A)	Control Room Em. Filter Unit North	Yes
3HV-B201A (HVC-203A)	Control Room Em. Filter Unit South	Yes

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
3HV-B198A (HVC-202A)	Control Room Em. Filter Unit North	Yes
3HV-B203A (HVC-204A)	Control Room Em. Filter Unit South	Yes
2SI-V327A (SI-407A)	RCS Loop 2 Shutdown Cooling Isolation	No
2SI-FM318A (SI-415A)	Shutdown Cooling Flow Control	No
2SI-V809A (SI-121A)	SI Pumps A Min. Flow Isol.	Yes
2HV-F253A (CAR-201A)	CARS Suction	Yes
2SI-V1549A1 (SI-139B)	LPSI Flow Control	Yes
2SI-V1539B1 (SI-138B)	LPSI Flow Control	Yes
2SI-V1545B1 (SI-225B)	HPSI Flow Control	Yes
2SI-V1547B3 (SI-227B)	HPSI Flow Control	Yes
3CH-V107B (BAM113B)	Boric Acid Gravity Feed	Yes
2SI-V802B (SI-120B)	SI Pumps B Min. Flow Isol.	Yes
2SI-V1540B2 (SI-226B)	HPSI Flow Control	Yes
1SI-V1502B (SI-401B)	RCS Loop 1 Shutdown Cooling Isolation	No
1SI-V1506TK1B (SI-331B)	Safety Inj. Tank 1B Isolation	Yes
1SI-V1508TK2B (SI-332B)	Safety Inj. Tank 2B Isolation	Yes
2HV-B159B (SBV-110B)	SBVS B Train Outlet	Yes
2HV-B161B (SBV-101B)	SBVS B Train Inlet	Yes
2HV-B163B (SBV-114B)	SBVS B Exhaust	Yes
2HV-B165B (SBV-113B)	SBVS B Recirc.	Yes

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
3HV-B197B (HVC-201B)	Control Room Em. Filter Unit North	Yes
3HV-B200B (HVC-203B)	Control Room Em. Filter Unit South	Yes
2CH-V123A/B (CVC-183)	Volume Control Tank Disch.	Yes
3HV-B199B (HVC-202B)	Control Room Em. Filter Unit North	Yes
3HV-B202B (HVC-204B)	Control Room Em. Filter Unit South	Yes
2SI-V326B (SI-407B)	RCS Loop 1 Shutdown Cooling Isolation	No
2SI-FM349B (SI-415B)	Shutdown Cooling Flow Control	No
2SI-V1544B4 (SI-228B)	HPSI Flow Control	Yes
2HV-F254B (CAR-201B)	CARS Suction	Yes
2SI-V810A (SI-120A)	S.I. Pumps A Min. Flow Isol.	Yes
3CH-V106A (BAM-113A)	Boric Acid Gravity Feed	Yes
2SI-V801-B (SI-121B)	S.I. Pumps B Min. Flow Isol.	Yes
2HV-B167A (CAR-204A)	CARS Disch.	Yes
3HV-B206A (HVR-313A)	CVAS A Train Outlet	Yes
3HV-B208A (HVR-304A)	CVAS A Train Inlet	Yes
2MS-V670 (MS-120A)	Steam Line 1 Upstream Normal Drain	No
2MS-V671 (MS-119A)	Steam Line 1 Upstream Emerg. Drain	No

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
2SI-V1534 (SI-219A)	HPSI Hdr. A Orifice Bypass	No
2SI-V1556 (SI-506A)	Hot Leg Injection	No
2SI-V1557 (SI-502A)	Hot Leg Injection	No
2HV-B168B (CAR-204B)	CARS Disch.	Yes
3HV-B207B (HVR-313B)	CVAS B Train Outlet	Yes
3HV-B209B (HVR-304B)	CVAS B Train Inlet	Yes
2MS-V663 (MS-120B)	Steam Line 2 Upstream Normal Drain	No
2MS-V664 (MS-119B)	Steam Line 2 Upstream Emerg. Drain	No
2SI-V811B (SI-219B)	HPSI Hdr. B Orifice Bypass	No
2SI-V1558 (SI-502B)	Hot Leg Injection	No
2SI-V1559 (SI-506B)	Hot Leg Injection	No
- (MS-416)	Emerg. Feed Water Pump Turbine Stop	No
1SI-V1501B (SI-405B)	Hyd. Pump Motor RCS Loop 1 Shutdown Cooling Isolation	No
1SI-V1503A (SI-405A)	Hyd. Pump Motor RCS Loop 2 Shutdown Cooling Isolation	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 following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 1720 ppm.

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 40 gpm of a solution containing at least 1720 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 1720 ppm, whichever is the more restrictive.

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 CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

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.

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

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.

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 within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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 purge 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.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be verified to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 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.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

REFUELING OPERATIONS

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3200 pounds, and an overload cut off limit of less than or equal to 3350 pounds for the fuel mast.

b. A minimum capacity of 1600 pounds and an overload cut off limit of less than or equal to 1700 pounds for the CEA mast.

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

ACTION:

a. With the above requirements for the fuel mast not satisfied, suspend use of the fuel mast from operations involving the movement of fuel assemblies.

b. With the above requirements for the CEA mast not satisfied, suspend use of the CEA mast from operations involving the movement of CEAs.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The fuel mast used for movement of fuel assemblies shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3200 pounds and demonstrating an automatic load cut off when the fuel mast load exceeds 3350 pounds.

4.9.6.2 The CEA mast used for movement of CEAs shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 1600 pounds and demonstrating an automatic load cut off when the CEA mast exceeds 1700 pounds.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Cranes in the fuel handling building shall be restricted as follows:

a. The spent fuel handling machine shall be used for the movement of fuel assemblies (with or without CEAs) and shall be OPERABLE with:

1. A minimum hoist capacity of 1800 pounds, and

2. An overload cutoff limit of less than or equal to 1900 pounds, and,

b. Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling building, or with fuel assemblies in the spent fuel pool.

ACTION:

a. With the spent fuel handling machine inoperable, suspend the use of the spent fuel handling machine for movement of fuel assemblies and place the crane load in a safe position.

b. With loads in excess of 2000 pounds over fuel assemblies in the spent fuel pool, place the crane load in a safe position.

SURVEILLANCE REQUIREMENTS

4.9.7.1 The spent fuel handling machine shall be demonstrated OPERABLE within 72 hours prior to the start of irradiated fuel assembly movement and at least once per 7 days thereafter by performing a load test of at least 1800 pounds and demonstrating tho automatic load cutoff when the hoist load exceeds 1900 pounds.

4.9.7.2 The electrical interlock system which prevents crane main hook travel over fuel assemblies in the spent fuel pool shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

4.9.7.3 Administrative controls which prevent crane auxiliary hook travel with loads in excess of 2000 pounds over the fuel assemblies in the spent fuel pool shall be enforced during crane operations.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling train shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling train OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling train to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm** at least once per 12 hours.

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*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

**The minimum flow may be reduced to 3000 gpm after the reactor has been shut down for greater than or equal to 175 hours or by verifying at least once per hour that the RCS temperature is less then 135°F.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two# independent shutdown cooling trains shall be OPERABLE and at least one shutdown cooling train shall be in operation.*

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 shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required trains to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling train 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 shutdown cooling train to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling train shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm** at least once per 12 hours.

#Only one shutdown cooling train is required to be OPERABLE provided there are no irradiated fuel assemblies seated within the reactor pressure vessel.

*The shutdown cooling loop may be removed from operations for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

**The minimum flow may be reduced to 3000 gpm after the reactor has been shut down for greater than or equal to 175 hours or by verifying at least once per hour that the RCS temperature is less than 135°F.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge valve isolation system shall be OPERABLE.

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

ACTION:

With the containment purge valve isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge valve isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge valve isolation occurs on manual initiation and on a containment purge isolation test signal from each of the radiation monitoring instrumentation channels.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.

CEAs

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the fuel seated in the reactor pressure vessel.

APPLICABILITY: During movement of CEAs within the reactor pressure vessel, when the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of CEAs.

REFEUELING OPERATIONS

3/4.9.11 WATER LEVEL - SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building ventilation systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool.

ACTION:

a. With one fuel handling building ventilation system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE fuel handling building ventilation system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no fuel handling building ventilation system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until at least one fuel handling building ventilation system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling building ventilation systems 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 filters and charcoal adsorbers and verifying that the system operates for at least 10 hours continuous 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:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the ventilation 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, and the system flow rate is 4000 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 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by 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.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.8 inches water gauge while operating the system at a flow rate of 4000 cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and draws flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the spent fuel pool area at a negative pressure of greater than or equal to 1/2-inch water gauge relative to the outside atmosphere during system operation.

4. Verifying that the filter cooling bypass valves can be manually cycled.

5. Verifying that the heaters dissipate 25 ± 1.3 kW when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

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 CEA worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 AND 3*.

ACTION:

a. With any full-length CEA 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 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 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 and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

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

*Operation in MODE 3 shall be limited to 6 consecutive hours.

SPECIAL TEST EXCEPTIONS

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient, and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4 10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.10.5 NATURAL CIRCULATION TESTING

LIMITING CONDITION FOR OPERATION

3.10.5 The limitation of Specification 3.4.1.2 may be suspended during the performance of natural circulation testing, provided the Reactor Coolant System saturation margin is maintained greater than or equal to 20°F.

APPLICABILITY: MODE 3 during natural circulation testing.

ACTION:

With the Reactor Coolant System saturation margin less than 20°F, immediately place at least one reactor coolant loop in operation, with at least one reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The saturation margin shall be determined to be within the above limits by continuous monitoring with the saturation margin monitors required by Table 3.3-10 or, by calculating the saturation margin at least once per 30 minutes.

4.10.5.2 The saturation margin monitor shall be demonstrated OPERABLE by performance of a CHANNEL CHECK within 24 hours prior to initiating natural circulation testing.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)a (μCi/mL)
A. Batch Waste Release Tanks[b,f,g,h,i]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
1. Boric Acid Condensate			I-131	1×10^{-6}
2. Waste Condensate	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
3. Laundry Waste	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
4. Turbine Building Industrial Waste Sumps*	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
5. Dry Cooling Tower Sumps #1 and #2*				
6. Regenerative Waste				
7. Filter Flush				
8. Waste				

*When release from this source is batch in nature.

TABLE 4.11-1 (Continued)

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)a (μCi/mL)
B. Continuous Releases,e,f,h,i	W Grab Sample	W	Principal Gamma Emitters[c]	$5x10^{-7}$
1. Turbine Building Industrial Waste Sumps**			I-131	$1x10^{-6}$
2. Dry Cooling Tower Sump #1**	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
3. Dry Cooling Tower Sump #2**	W Grab Sample	M Composite[d]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
4. Circulating Water Discharge-Steam Generator Blow-down HX	W Grab Sample	Q Composite[d]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
5. Auxiliary Component Cooling Water Pumps				

**When release from this source is continuous in nature.

TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[d]A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

[e]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

[f]Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

[g]If the contents of the filter flush tank or the regenerative waste tank contain detectable radioactivity, no discharges from these tanks shall be made to the UNRESTRICTED AREA and the contents of these tanks shall be directed to the liquid radwaste treatment system.

[h]Turbine Building Industrial Waste Sump (TBIWS)

The TBIWS shall be required to be sampled and analyzed in accordance with this table if any of the following conditions exist:

(1) Primary to secondary leakage is occurring; or,

(2) Activity is present in the secondary system as indicated by either the SGB monitors or secondary sampling and analysis; or,

(3) Activity was present in the TBIWS during the previous 4 weeks.

If none of the above situations exists, then the sampling and analysis of this stream need not be performed.

[i]Sampling and analysis of the dry cooling tower sumps and the auxiliary component cooling water pump discharge will be required only when detectable activity exists in the CCW.

Sampling and analysis of the circulating water discharge-steam generator blowdown heat exchanger discharge (CWD-SGB) will be required only when detectable activity exists in the secondary system.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent to UNRESTRICTED AREAS (see Figure 5.1-3) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the liquid radwaste treatment system not in operation, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be demonstrated OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 1.57×10^{-2} curies,* excluding tritium and dissolved or entrained noble gases. For outside temporary storage tanks, the curie content shall be limited such that a rupture will not result in exceeding 10 CFR Part 20 limits at the UNRESTRICTED AREA boundary.

a. PWST
b. Outside temporary tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Based on 80% tank capacity and an MPC_ℓ of 2×10^{-5} µCi/mL Cesium-137 equivalent, 10 CFR Part 20, Appendix B, Table II.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s), and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 Representative samples and analysis of gaseous effluents shall be obtained in accordance with the sampling and analyses program specified in Table 4.11-2.

4.11.2.1.3 Based upon the sampling and analysis performed in Table 4.11-2 the dose rate due to I-131, I-133, H-3, and all other radionuclides in particulate form with half-lives greater than 8 days shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/mL)
A. Waste Gas Holdup Tanks	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE (Plant Stack)	P Each PURGE[c] Grab Sample	P Each PURGE[c]	Principal Gamma Emitters[b]	1×10^{-4}
		M	H-3	1×10^{-6}
C.1 Plant Stack	M[c,d,i] Grab Sample	M	Principal Noble Gas Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C.2 Fuel Handling Building Ventilation (Normal) Exhaust	M[e,j] Grab Sample	M	Principal Noble Gas Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
D.1 All Release Types as listed in B., C.1, and C.2 above	Continuous[f,h,j]	W[g] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
D.2 Main Condenser Evacuation and Turbine Gland Sealing System	Continuous[f,h,j]	W[g] Particulate Sample	Principal Particulate Gamma Emitters[b]	1×10^{-11}
	Continuous[f,h,j]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f,h,j]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f,h,j]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATIONS

[b]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, I-133, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[c]Sampling shall also be performed within 24 hours following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period. Analysis for principle gamma emitters as defined in (b) above shall be completed within 48 hours of sampling.

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

[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]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 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

[h]If no primary to secondary leakage exists, then only the gross beta or gamma noble gases analysis need be performed for the main condenser evacuation and turbine gland sealing system. If a primary to secondary leak exists and the release from the main condenser evacuation and turbine gland sealing system has not been released via the plant stack, then the sampling and analysis must be performed.

[i]Note (c) above is not applicable for the plant stack unless the noble gas monitor shows that effluent activity has increased by a factor of 3.

[j]Fuel Handling Building sampling is required whenever irradiated fuel is in the storage pool.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed either:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.2.4.2 The installed Gaseous Radwaste Treatment System, shall be demonstrated OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

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 the WASTE GAS HOLDUP SYSTEM 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 the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and immediately reduce the concentration of oxygen to less than or equal to 4% by volume and then take the ACTION in a. above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 8.5×10^4 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank on-service shall be determined to be within the above limit at least once per 7 days until the quantity exceeds 4.25×10^4 curies noble gases (50% of allowed limit) and then at least once per 24 hours when radioactive materials are being added to the tank. Tanks isolated for decay will be sampled to verify above limit is met within 24 hours following removal from service.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be SOLIDIFIED or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the solid waste system as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and (2) take appropriate administrative action to prevent recurrence.

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to relases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

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 Specification 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, calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 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 methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4.a.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specificaton 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. DIRECT RADIATION[b]	31 routine monitoring stations either with 2 or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY, an outer ring of stations, 1 in 10 of the meteorological sectors in the 6- to 8-km range from the site; the balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.

*The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sampling program.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. AIRBORNE Radioiodine and Particulates	Samples from 5 locations 3 samples from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q. 1 sample from the vicinity of a community having the highest calculated annual average ground-level D/Q. 1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
3. WATERBORNE			
a. Surface[f]	1 sample upstream 1 sample downstream	Composite sample over 1-month period[g]	Gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources only if likely to be affected[h]	Quarterly	Gamma isotopic[e] and tritium analysis quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
c. Drinking	1 sample of each of 1 to 3 of the nearest water supplies that could be affected by its discharge.	Composite sample over 2-week period[g] when I-131 analysis is performed, monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[i] Composite for gross beta and gamma isotopic analyses[e] monthly. Composite for tritium analysis quarterly.
d. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr[i]. 1 sample from milking animals at a control location 15-30 km distant and in the least prevalent wind direction.[c]	Semimonthly when animals are on pasture; monthly at other times	Gamma isotopic[e] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
b. Fish and Invertebrates	1 sample of each commercially and recreationally important species in vicinity of plant discharge area.	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis[e] on edible portions.
	1 sample of same species in areas not influenced by plant discharge.		
c. Food Products	1 sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have discharged.	At time of harvest[j]	Gamma isotopic analysis[e] on edible portion.
	Samples of 1 to 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available	Gamma isotopic[e] and I-131 analysis.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, corrective action shall be completed prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semianuual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

[c]The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

[d]Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

[e]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

[f]The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

[g]A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

[h]Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[i]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[j]If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b,c]

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GAS (pCi/m^3)	FISH (pCi/kg,wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg,wet)	SEDIMENT (pCi/kg,dry)
gross beta	4	0.01				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

TABLE 4.12-1 (Continued)

TABLE NOTATIONS

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta T)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

[d] LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8 identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

*Broad leaf vegetation sampling of different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1 Part 4.c. shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two containment spray systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required containment spray systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment, or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems, or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems, or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 5.15% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from any postulated accident are minimal and a 2% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The Surveillance Requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 520°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature, and (5) the ECCS analysis remains valid for the peak linear heat rate of Specification 3.2.1.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 2.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions assuming the most reactive CEA stuck out of the core and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 53,494 gallons of 1720 ppm borated water from the refueling water storage pool. The higher limit of 447,100 gallons is specified to be consistent with Specification 3.5.4 in order to meet the ECCS requirements.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5,465 gallons of 1720 ppm borated water from the refueling water storage pool or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

BASES

BORATION SYSTEMS (Continued)

The contained water volume limits include allowance for water not available because of discharge line location, instrument tolerances, and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The lower limit on the contained water volume, the specified boron concentration, and the physical size (approximately 600,000 gallons) of the RWSP also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum limit on the RWSP temperature ensures that the assumptions used in the containment pressure analysis under design base accident conditions remain valid and avoids the possibility of containment overpressure. The minimum limit on the RWSP temperature is required to prevent freezing and/or boron precipitation in the RWSP.

3/4.1.2.9 BORON DILUTION

This specification is provided to prevent a boron dilution event, and to prevent a loss of SHUTDOWN MARGIN should an inadvertent boron dilution event occur. Due to boron concentration requirements for the RWSP and boric acid makeup tanks, the only possible boron dilution that would remain undetected by the operator occurs from the primary makeup water through the CVCS system. Isolating this potential dilution path or the OPERABILITY of the startup channel high neutron flux alarms, which alert the operator with sufficient time available to take corrective action, ensures that no loss of SHUTDOWN MARGIN and unanticipated criticality occur. The requirement to remove power on two charging pumps in MODE 5 with the reactor coolant loops drained is necessary because there is insufficient time for operator response in the event two or three charging pumps inject unborated water into the RCS in this condition.

The ACTION requirements specified in the event startup channel high neutron flux alarms are inoperable provide an alternate means to detect boron dilution by monitoring the RCS boron concentration to detect any changes. The frequencies specified in Table 3.1-1 provide the operator sufficient time to recognize a decrease in boron concentration and take appropriate corrective action without loss of SHUTDOWN MARGIN. More frequent checks are required with more charging pumps in operation due to the higher potential boron dilution rate.

The surveillance requirements specified provide assurance that the startup channel high neutron flux alarms remain OPERABLE and that required valve and electrical lineups remain in effect.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is

MOVABLE CONTROL ASSEMBLIES (Continued)

maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is (1) a small effect on the time dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit

MOVABLE CONTROL ASSEMBLIES (Continued)

continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 520°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The establishment of LSSS and LCOs requires that the expected long and short-term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady-state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accommodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accommodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that (1) the minimum SHUTDOWN MARGIN is maintained, and (2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long-term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

The Part Length CEA Insertion Limits of Specification 3.1.3.7 ensure that adverse power shapes and rapid local power changes which affect radial peaking factors and DNB considerations do not occur as a result of a part-length CEA group covering the same axial segment of the fuel assemblies for an extended period of time during operation.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provides adequate monitoring of the core power distribution and is capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limit of 13.4 kW/ft is not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady-state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_r measurement uncertainty factor of 1.080, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for flux peaking augmentation and rod bow.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs (assuming minimum core power of 20% of RATED THERMAL POWER). The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-3 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

These penalty factors are determined from uncertainties associated with planar radial peaking measurements, engineering heat flux uncertainty, axial densification, software algorithm modelling, computer processing, rod bow, and core power measurement.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic Surveillance Requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provide assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The Surveillance Requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady-state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q \, g \cos (\theta - \theta_o)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

AZIMUTHAL POWER TILT - T_q (Continued)

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provides adequate monitoring of the core power distribution and is capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide a 95/95 probability/confidence level that the core power calculated by COLSS, based on the minimum DNBR limit, is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurements, engineering heat flux, state parameter measurement, software algorithm modelling, computer processing, rod bow, and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors plus those associated with startup test acceptance criteria are also included in the CPCs which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

DNBR MARGIN (Continued)

The DNBR penalty factors listed in Specification 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses, and that the DNBR is maintained within the safety limit for Anticipated Operational Occurrences (AOO).

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses, with adjustment for instrument accuracy of $\pm 2^{o}F$, and that the peak linear heat generation rate and the moderator temperature coefficient effects are validated.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses, to ensure that the peak linear heat rate and DNBR remain within the safety limits for Anticipated Operational Occurrences (AOO).

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses. The inputs to CPCs and COLSS are the most limiting. The values are adjusted for an instrument accuracy of ± 25 psi. The sensitive events are SGTR, LOCA, FWLB and loss of condenser vacuum to initial high pressure, and MSLB to initial low pressure.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the Reactor Protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The redundancy design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs will use DNBR and LPD penalty factors to restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded, a reactor trip will occur.

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." Table 3.3-10 includes Regulatory Guide 1.97 Category I key variables. The remaining Category I variables are included in their respective specifications.

3/4.3.3.7 CHEMICAL DETECTION SYSTEMS

The OPERABILITY of the chemical detection systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chemical release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975 and the recommendations of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," June 1974.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment, or structures.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 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, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops or trains (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling (low pressure safety injection) 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 reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to 285°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety

SAFETY VALVES (Continued)

valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the overpressure protection system provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached and also assuming no operation of the steam dump valves.

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

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized while uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an SIAS test signal the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4.4 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

STEAM GENERATORS (Continued)

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.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 gpm per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

If one of the required systems becomes inoperable, 30 days are permitted for restoration since two diverse and redundant RCS leakage detection systems remain OPERABLE. If, however, the inoperable system is the containment gaseous or particulate monitoring system, grab samples are also performed as a backup to the single remaining atmospheric monitoring system.

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm leakage limit per steam generator (720 gal/day) ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady-state primary-to-secondary steam generator leakage rate of 1 gpm and a concurrent loss-of-offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Waterford Unit 3 site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6-month consecutive period with greater than 1 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figures 3.4-2 and 3.4-3. The limitations on the Reactor Coolant System heatup and cooldown rates are further restricted due to stress limitations in the Reactor Coolant Pump. As part of the LOCA support scheme, the Reactor Coolant Pump has a ring around the suction nozzle of the pump. The support skirt is welded to the ring. Due to this design, the heatup and cooldown rates must be limited to maintain acceptable thermal stresses.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using FSAR Table 5.3-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50.

TABLE B 3/4.4-1

REACTOR VESSEL FRACTURE TOUGHNESS

Piece Number	Drawing Number	Code Number	Material	Vessel Location	Drop Weight NDTT(F)	$RT_{NDT}^{(A)}$(F)	Charpy 30 ft-lb. Fix Temp. (F) Long.	Charpy 50 ft-lb. Fix Temp. (F) Long.	35 Mils Lateral Expansion Temp. (F) Long.	Charpy Upper Shelf Energy (ft-lb.) Long.
106-101	741701 6103	M-603-1	SA508 CL-II	Closure Head Flange	20	20	-15	2	-5	159
126-101	741701 6103	M-1001-1	SA508 CL-II	Vessel Flange	20	20	-20	-5	2	154
131-102A	741701 6103	M-1013-1	SA508 CL-I	Safe End	0	0	-35	0	-15	148
131-102D	741701 6103	M-1013-2	SA508 CL-I	Safe End	0	0	-35	0	-15	148
131-102C	741701 6103	M-1013-3	SA508 CL-I	Safe End	0	0	-27	0	-32	149
131-102B	741701 6103	M-1013-4	SA508 CL-I	Safe End	0	0	-27	0	-30	149
131-101A	741701 6103	M-1012-1	SA508 CL-I	Safe End	0	0	0	25	-5	146
131-101B	741701 6103	M-1012-2	SA508 CL-I	Safe End	0	0	0	25	-5	146
128-301	741701 6103	M-1011-1	SA508 CL-II	Outlet Nozzle	-20	-20	-37	0	5	99
128-101	741701 6103	M-1010-1	SA508 CL-II	Inlet Nozzle	20	20	-37	10	0	135
128-101	741701 6103	M-1010-2	SA508 CL-II	Inlet Nozzle	20	20	-50	-35	-40	140
128-101	741701 6103	M-1010-3	SA508 CL-II	Inlet Nozzle	10	10	-70	-47	-42	133
128-101	741701 6103	M-1010-4	SA508 CL-II	Inlet Nozzle	30	30	-40	-20	-30	140
128-301	741701 6103	M-1011-2	SA508 CL-II	Outlet Nozzle	0	0	-30	-10	-12	188
124-102	741701 6103	M-1003-1	SA533-B CL-I	Intermediate Shell Plate	-30	-30	-30	-10	-10	144
124-102	741701 6103	M-1003-2	SA533-B CL-I	Intermediate Shell Plate	-50	-50	-55	-12	-15	149

TABLE B 3/4.4-1 (Continued)

Piece Number	Drawing Number	Code Number	Material	Vessel Location	Drop Weight NDTT(F)	RT_{NDT}(F)(A)	Charpy 30 ft-lb. Fix Temp. (F) Long.	Charpy 50 ft-lb. Fix Temp. (F) Long.	35 Mils Lateral Expansion Temp. (F) Long.	Charpy Upper Shelf Energy (ft-lb.) Long.
124-102	741701 6103	M-1003-3	SA533-B CL-I	Intermediate Shell Plate	-50	-42	-22	-2	-10	138
122-102	741701 6103	M-1002-1	SA533-B CL-I	Upper Shell Plate	-40	-8	13	32	23	151
122-102	741701 6103	M-1002-2	SA533-B CL-I	Upper Shell Plate	-20	-20	-20	12	15	128
122-102	741701 6103	M-1002-3	SA533-B CL-I	Upper Shell Plate	-40	-40	-20	0	0	153
154-102	741701 6103	M-1007-1	SA533-B CL-I	Bottom Head Torus	-80	-80	-72	-62	-60	174
152-101	741701 6103	M-1008-1	SA533-B CL-I	Bottom Head Dome	-40	-40	-35	-10	-15	141
104-102	741701 6103	M-1005-1	SA533-B CL-I	Closure Head Torus	-30	-30	-25	0	-2	160
142-101	741701 6103	M-1004-1	SA533-B CL-I	Lower Shell Plate	-50	-15	10	25	20	163
142-101	741701 6103	M-1004-2	SA533-B CL-I	Lower Shell Plate	-20	22	37	62	55	144
142-101	741701 6103	M-1004-3	SA533-B CL-I	Lower Shell Plate	-50	-10	12	30	25	145
102-101	741701 6103	M-1006-1	SA533-B	Closure Head Dome	-50	-25	-5	15	10	138

(A) MTEB Position 5-2 "Fracture Toughness Requirements," Paragraph 1.1(3)(b).

PRESSURE/TEMPERATURE LIMITS (Continued)

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia (as corrected for elevation and instrument error).

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of the shutdown cooling system relief valve or an RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 285°F. Each shutdown cooling system relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with injection into a water-solid RCS. The limiting transient includes simultaneous, inadvertent operation of three HPSI pumps, three charging pumps, and all pressurizer backup heaters in operation. Since SIAS starts only two HPSI pumps, a 20% margin is realized.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to 285°F, are provided to prevent RCS pressure transients caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures. Maintaining the steam generator less than 100°F above each of the Reactor Coolant System cold leg temperatures (even with the RCS filled solid) or maintaining a large surge volume in the pressurizer ensures that this transient is less severe than the limiting transient considered above.

The automatic isolation setpoint of the shutdown cooling isolation valves is sufficiently high to preclude inadvertent isolation of the shutdown cooling relief valves during a pressure transient.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

The ACTION requirements provide restrictions prior to restoring structural integrity on ASME Code Class 1 and 2 components while allowing sufficient Reactor Coolant System heatup to allow pressurization to perform hydrostatic testing of the affected component while complying with the RCS pressure/ temperature limits of Specification 3.4.8.1.

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1980 Edition and Addenda through Winter 1981.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the assumptions used for safety injection tank injection in the safety analysis are met.

The safety injection tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provides this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of water borated within RWSP boron concentration limits provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

The requirement to verify the minimum pump discharge pressure on recirculation flow ensures that the pump performance curve has not degraded below that used to show that the pump exceeds the design flow condition assumed in the safety analysis and is consistent with the requirements of ASME Section XI.

3/4.5.4 REFUELING WATER STORAGE POOL (RWSP)

The OPERABILITY of the refueling water storage pool (RWSP) as part of the ECCS also ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWSP minimum volume and boron concentration ensure that (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWSP and the RCS water volumes with all CEAs inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

REFUELING WATER STORAGE POOL (Continued)

The contained water volume limit includes an allowance for water not usable because of pool discharge line location or other physical characteristics.

The lower limit on contained water volume, the specific boron concentration and the physical size (approximately 600,000 gallons) of the RWSP also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The maximum limit on the RWSP temperature ensures that the assumptions used in the containment pressure analysis under design base accident conditions remain valid and avoids the possibility of containment overpressure. The minimum limit on the RWSP temperature is required to prevent freezing and/or boron precipitation in the RWSP.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance requirements for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.65 psid, (2) the containment peak pressure does not exceed the design pressure of 44 psig during either LOCA or steam line break conditions, and (3) the minimum pressure of the ECCS performance analysis (BTP CSB 6-1) is satisfied.

The maximum peak pressure expected to be obtained from a steam line break event is 43.76 psig. The limits of Figure 3.6-1 for initial positive containment pressure and temperature will limit the total pressure to less than 44 psig which is less than the design pressure and is consistent with the safety analyses.

The limit of 14.375 psia for initial negative containment pressure ensures that the minimum containment pressure is consistent with the ECCS performance analysis ensuring core reflood under LOCA conditions.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 269.3°F during LOCA conditions and 413.5°F during MSLB conditions and is consistent with the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 43.76 psig in the event of a main steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The use of the containment purge valves is restricted to 90 hours per year in accordance with Standard Review Plan 6.2.4 for plants with the Safety Evaluation Report for the Construction License issued prior to July 1, 1975. The purge valves have been modified to limit the opening to approximately 52° to ensure the valves will close during a LOCA or MSLB; and therefore, the SITE BOUNDARY doses are maintained within the guidelines of 10 CFR Part 100. The purge valves, as modified, comply with all provisions of BTP CSB 6-4 except for the recommended size of the purge line for systems to be used during plant operation.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM (Continued)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT SPRAY SYSTEM and CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Spray System and the Containment Cooling System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or MSLB for any double-ended break of the largest reactor coolant pipe or main steam line. Under post-accident conditions these systems will maintain the containment pressure below 44 psig and temperatures below 269.3°F during LOCA conditions or 413.5°F during MSLB conditions. The systems also reduce the containment pressure by a factor of 2 from its post-accident peak within 24 hours, resulting in lower containment leakage rates and lower offsite dose rates.

The Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere under post-LOCA conditions to maintain doses in accordance with 10 CFR Part 100 limits as described in Section 6.5.2 of the FSAR.

When shutdown cooling is placed in operation, the Containment Spray System is no longer required OPERABLE in order to allow realignment and isolation of the spray headers. This is necessary to avoid a single failure of the spray header isolation valve causing Reactor Coolant System depressurization and inadvertent spraying of the containment. At the reduced RCS pressure and temperature associated with entry into shutdown cooling, the probability and associated heat loads of a LOCA or MSLB are greatly reduced. The OPERABILITY of the Containment Cooling System in MODE 4 is sufficient to provide depressurization and cooling capability.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

3/4.6.5 VACUUM RELIEF VALVES

The OPERABILITY of the primary containment to annulus vacuum relief valves with a setpoint of less than or equal + 0.3 psid ensures that the containment internal pressure differential does not become more negative than the containment design limit for internal pressure differential of 0.65 psi. This situation would occur, for the worst case, if all containment heat removal systems (containment spray, containment cooling, and other HVAC systems) were inadvertently started with only one vacuum relief valve OPERABLE.

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during LOCA conditions.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52.

3/4.6.6.2 SHIELD BUILDING INTEGRITY

SHIELD BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the shield building ventilation system, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.6.3 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient demonstrate this capability.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1210 psia) of its design pressure of 1100 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 16.267×10^6 lb/hr (at 1210 psia design pressure less 50 psi pressure drop to the inlet of the valves) which is 104.2% of the total secondary steam flow plus 2% uncertainties of 15.61×10^6 lb/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two-loop 4 pump operation

$$SP = \frac{6-N}{6} \times 104.2\%$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

N = maximum number of inoperable safety valves per steam line

104.2 = the ratio of the total relieving capacity of all 12 main steam safety valves divided by the secondary steam flow at 100% Rated Thermal Load, plus 2% uncertainty.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

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

Each electric-driven emergency feedwater pump is capable of delivering a total feedwater flow of 350 gpm at a pressure of 1163 psig to the entrance of the steam generators. The steam-driven emergency feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1163 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

The surveillance requirement to verify the minimum pump discharge pressure on recirculation flow ensures that the pump performance curve has not degraded below that used to show that the pumps meet the above flow requirements and is consistent with the requirements of ASME Section XI.

3/4.7.1.3 CONDENSATE STORAGE POOL

The OPERABILITY of the condensate storage pool with the minimum water volume ensures that sufficient water is available to cool the Reactor Coolant System to shutdown cooling entry conditions (350°F) following any design basis accident. Additional makeup water is stored in the wet cooling tower basins providing the capability to maintain HOT STANDBY conditions for at least an additional 2 hours prior to initiating shutdown cooling. The total makeup capacity also provides sufficient cooling for 24 hours until shutdown cooling is initiated in the event the cooling towers sustain tornado damage concurrent with the accident or if natural circulation cooldown is required. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The combined capacity is sufficient to maintain the plant at HOT STANDBY for 4 hours, followed by a cooldown to shutdown cooling entry conditions assuming the availability of only onsite power or only offsite power, and the worst single failure (loss of a diesel generator or atmospheric dump valve). This requires approximately 275,000 gallons and complies with BTP RSB 5-1.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite 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 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss-of-offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down 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 safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator secondary pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitation to 115°F and 210 psig is based on a steam generator RT_{NDT} of 40°F and is sufficient to prevent brittle fracture. Below this temperature of 115°F the system pressure must be limited to a maximum of 20% of the secondary hydrostatic test pressure of 1375 psia (corrected for instrument error). The limitations on the primary side of the steam generator are bounded by the restrictions on the reactor coolant system in Specification 3.4.8.1.

3/4.7.3 COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEMS

The OPERABILITY of the component cooling water system and its corresponding auxiliary 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 these systems, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level, temperature, and number of fans ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.5 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 27.0 ft Mean Sea Level is based on the maximum elevation at which the levee provides protection, the nuclear plant island structure provides protection to safety-related equipment up to elevation +30 ft Mean Sea Level.

3/4.7.6 CONTROL ROOM AIR CONDITIONING SYSTEM

The OPERABILITY of the control room air conditioning system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analysis and is consistent with Regulatory Guide 1.52.

3/4.7.7 CONTROLLED VENTILATION AREA SYSTEM

The OPERABILITY of the controlled ventilation area system ensures that radioactive materials leaking from the penetration area or the ECCS equipment within the pump room 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 safety analyses.

CONTROLLED VENTILATION AREA SYSTEM (Continued)

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Operations Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as

SNUBBERS (Continued)

a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elasped (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

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 the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is 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 shield mechanism.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, Halon, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.11 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.12 ESSENTIAL SERVICES CHILLED WATER SYSTEM

The OPERABILITY of the essential services chilled water system ensures that sufficient chilled water is supplied to those air handling systems which cool spaces containing equipment required for safety-related operations and, during normal plant operation, the nonessential spaces.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.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 Part 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 A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

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, and 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 provision allowing diesel generator starts utilizing manufacturers' recommended prelube and/or warmup procedures, including longer starting and loading periods, is to minimize stress and wear on the diesel engine and is in accordance with Generic Letter 84-15 concerning Diesel Generator Reliability and Station Blackout. Fast starts from ambient conditions (includes lubricating and warmup systems operating while in standby lineup) at least once every 184 days is in accordance with RRAB PRA analysis of this surveillance.

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

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries 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 on float 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.

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 0.015 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 than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 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 0.020 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 0.040 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.

3/4.8.4 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.

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.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 1720 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of CEAs and fuel assemblies, (2) each hoist has sufficient load capacity to lift a CEA or fuel assembly, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - FUEL HANDLING BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA, and associated handling tool over other fuel assemblies in the spent fuel pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling train will not result in a complete loss of decay heat removal capability. When there is no irradiated fuel in the reactor pressure vessel, this is not a consideration and only one shutdown cooling train is required to be OPERABLE. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

The limitations on the fuel handling building ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. Obtaining and analyzing charcoal samples after 720 hours of adsorber operation (since the last sample and analysis) ensures that the adsorber maintains the efficiency assumed in the safety analyses and is consistent with Regulatory Guide 1.52.

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 MTC, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 NATURAL CIRCULATION TESTING

This special test exception permits all reactor coolant pumps to be secured during natural circulation testing and operator training for periods in excess of the 1 hour allowed by Specification 3.4.1.2.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration is water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The sampling and analysis of the contents of the regenerative waste tank and the filter flush tank is performed if primary to secondary leakage occurs in a steam generator. The contents of these tanks cannot be discharged to the UNRESTRICTED AREA if any radioactivity is detected in these tanks since the discharge from these tanks is unmonitored. When radioactivity is detected in these tanks, the contents from these tanks must be discharged to the liquid radwaste system where the contents may then be monitored upon discharge.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibilty and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation

DOSE (Continued)

of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirement in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment . The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of

DOSE - NOBLE GASES (Continued)

Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977.

The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

Grab sampling of effluents from the main condenser evacuation and turbine gland sealing system is not required when this source has been continously discharging to the plant stack over the past 30 days. If no primary to secondary leakage in the steam generator exists, then there should be no radioactive release from the main condenser evacuation and turbine gland sealing system and the gross beta or gamma monitoring for noble gases will be sufficient to determine if any radioactivity is present in the release. If a primary to secondary leak exists, then the release from the main condenser evacuation and turbine gland sealing system will be sampling will be sampled and analyzed in accordance with Table 4.11-2.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with

DOSE-IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The discharge from the main condenser evacuation and turbine gland sealing system shall be required to be directed to the plant stack when the release rate of I-131 from this source is $\geq 2 \times 10^{-4}$ μCi/s. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

This specification considers postulated radioactive releases due to a waste gas system leak or failure, and limits the quantity of radioactivity contained in each pressurized gas storage tank in the WASTE GAS HOLDUP SYSTEM to assure that a release would be substantially below the guidelines of 10 CFR Part 100 for a postulated event.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR 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, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 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 8 km must be considered. If the dose to

TOTAL DOSE (Continued)

any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1-3.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor building is a steel containment vessel surrounded by a reinforced concrete shield building.

5.2.1.1 The containment vessel is a cylindrical vessel with a hemispherical dome and an ellipsoidal bottom and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 240 feet 4.75 inches.

c. Nominal thickness of cylindrical wall = 1.903 inches.

d. Nominal thickness of hemispherical dome = 0.95 inch.

e. Nominal thickness of ellipsoidal bottom = 2.25 inches.

f. Net free volume = 2,680,000 cubic feet.

5.2.1.2 The shield building is a cylindrical structure with a shallow dome roof and having the following design features:

a. Nominal outside diameter = 154 feet.

b. Nominal outside height (from base slab) = 249.5 feet.

c. Nominal thickness of cylindrical wall = 3 feet.

d. Nominal thickness of dome roof = 2.5 feet.

e. Nominal inside radius of dome roof = 112 feet.

f. Nominal annular space = 4 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 263°F.

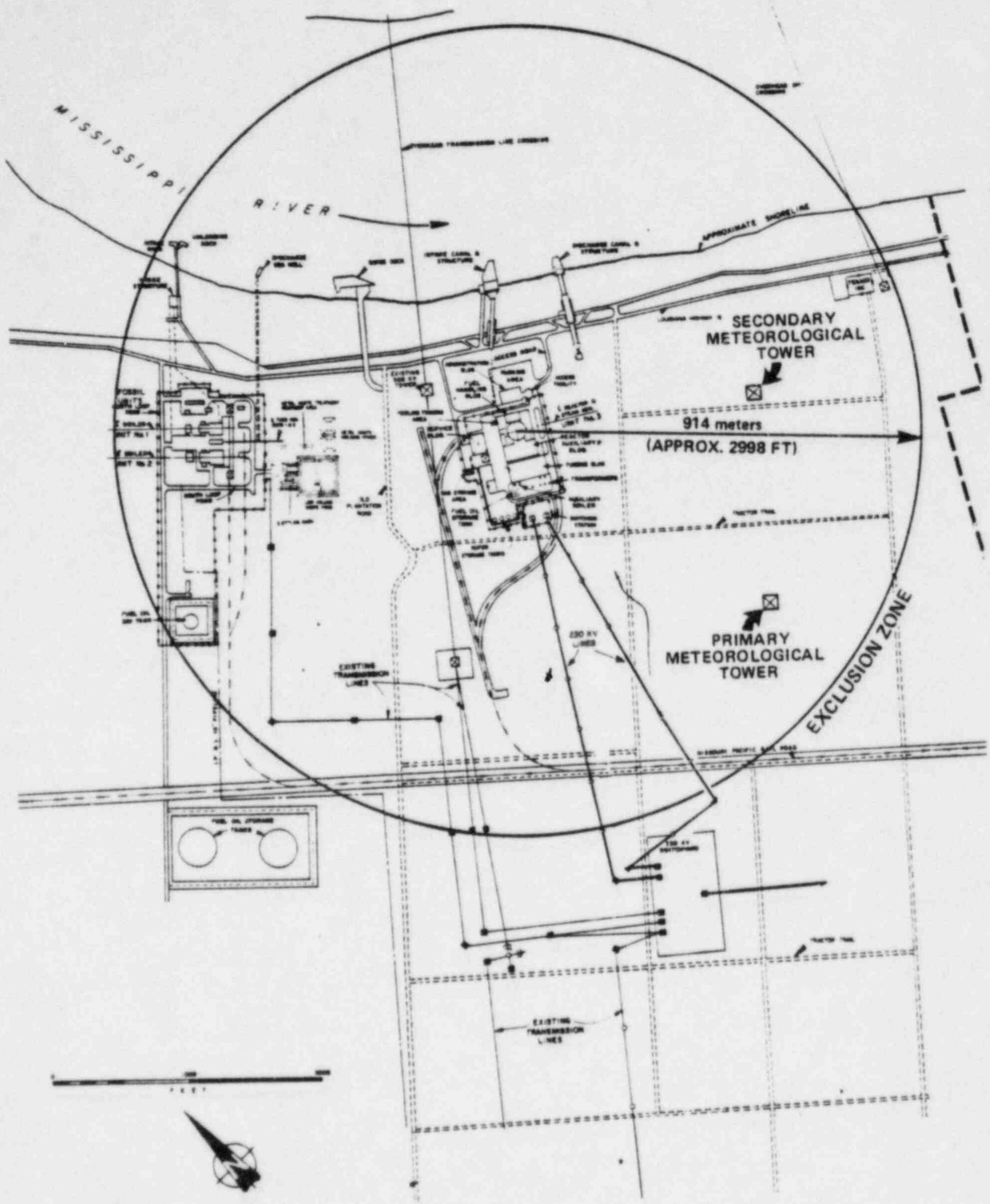

FIGURE 5.1-1

EXCLUSION AREA

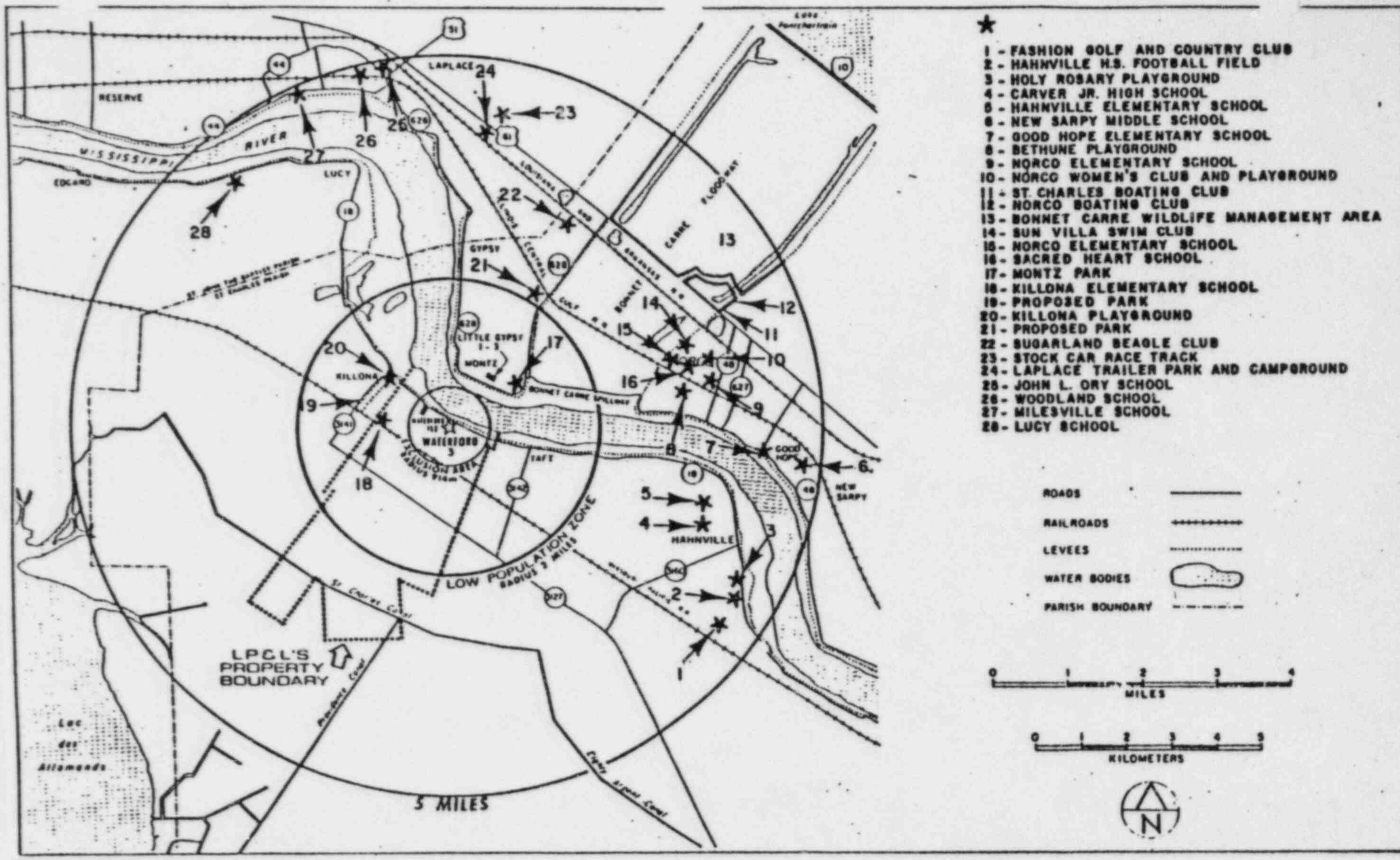

FIGURE 5.1-2 LOW POPULATION ZONE

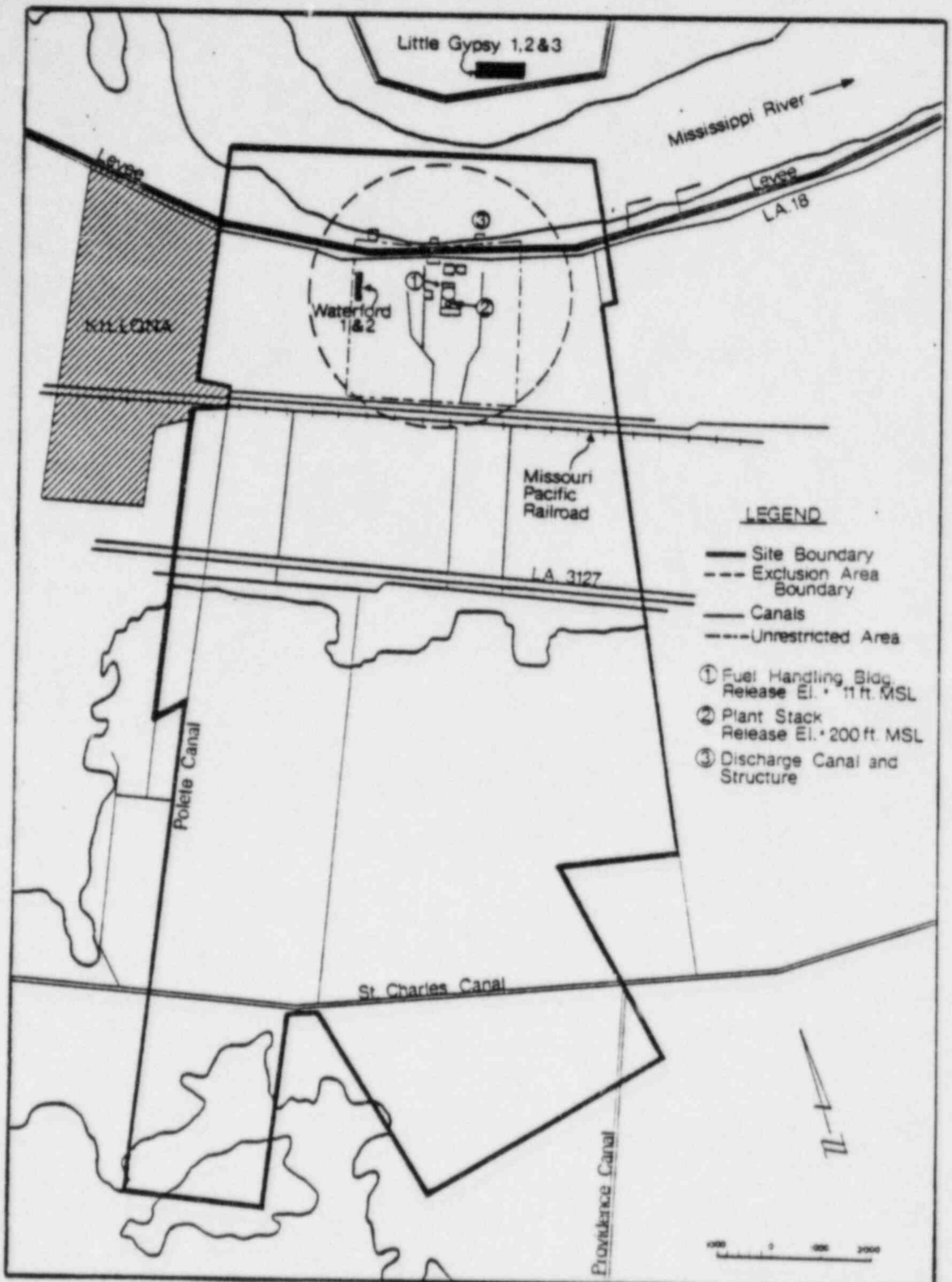

FIGURE 5.1-3 SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1807 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.70 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full-length and 8 part-length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer and surge line which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,800 +600, -0 cubic feet at a nominal T_{avg} of 582.1°F.

5.5 METEOROLOGICAL TOWERS LOCATION

5.5.1 The primary and backup meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 0.0455 delta k_{eff} for uncertainties as described in Section 9.1.2 and Table 9.1-8 of the FSAR.

b. A nominal 10.38 inch center-to-center distance between fuel assemblies placed in the spent fuel storage racks.

5.6.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation +40.0 MSL.

CAPACITY

5.6.4 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1088 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

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 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup cycles and 500 cooldown cycles at rates $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 70°F to $\geq$ 545°F; cooldown cycle - T_{avg} from $\geq$ 545°F to $\leq$ 70°F.
	500 pressurizer heatup and cooldown cycles at rates $\leq$ 200°F/h.	Heatup cycle - Pressurizer temperature from < 70°F to > 653°F; cooldown cycle $\geq$ 653°F to $\leq$ 70°F
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psia with RCS temperature $\geq$ 60°F above the most limiting components' NDTT value.
	200 leak testing cycles.	RCS pressurized to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than 400°F.
	200 seismic stress cycles.	Subjection to a seismic event equal to the operating basis earthquake (OBE).
	480 cycles (any combination) of reactor trip, turbine trip, or complete loss of forced reactor coolant flow	Trip from 100% of RATED THERMAL POWER; Turbine trip (total load rejection from 100% of RATED THERMAL POWER followed by resulting reactor trip; simultaneous loss of all reactor coolant pumps at 100% of RATED THERMAL POWER
	5 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODE 1, 2, or 3.

TABLE 5.7-1 (Continued)

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray Nozzle	Unlimited number of cycles	Main spray (4 pumps operating). Main spray (less than 4 pumps operating) with $\Delta T_m \leq 130°F$ Auxiliary spray with $\Delta T_a \leq 140°F$
	Calculate usage factor as indicated below:	Main spray (Less than 4 pumps operating) with $\Delta T_m > 130°F$ Auxiliary spray with $\Delta T_a > 140°F$

Calculational Method:

1. The spray cycle is defined as the opening and closing of a spray valve, either by main spray or auxiliary spray.

2. If the difference between pressurizer water temperature and the spray water temperature exceeds 130°F for main spray or 140°F for auxiliary spray, each spray cycle and the corresponding temperature difference is logged.

3. The spray nozzle usage factor shall be calculated as follows::

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   D. Add $\Sigma N/N_A$ for both main and auxiliary spray. This total is the cumulative usage factor.

4. A. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

   B. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of nozzle fatigue shall be performed within 90 days. The evaluation shall determine that the nozzle remains acceptable for additional service beyond the 90-day period or subsequent spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 130°F for main spray and 140°F for auxiliary spray.

TABLE 5.7-1 (Continued)

PRESSURIZER SPRAY NOZZLE USAGE FACTOR

MAIN SPRAY				AUXILIARY SPRAY			
ΔT_m	N_A	N	N/N_A	ΔT_a	N_A	N	N/N_A
130-180	7900			140-190	50000		
181-230	4500			191-240	2200		
231-280	2900			241-290	1300		
281-330	1900			291-340	850		
331-380	1200			341-390	550		
381-430	850			391-440	375		
431-480	555			441-490	225		
				491-540	150		
		$\Sigma N/N_A$=	________			$\Sigma N/N_A$=	________

Cumulative Usage Factor

$\Sigma N/N_A$ (Main Spray) ________

$\Sigma N/N_A$ (Aux. Spray) ________

Total ________ (Cumulative Usage Factor)

WHERE:

ΔT_a = Temperature Difference between pressurizer and auxiliary spray (charging line)

ΔT_m = Temperature Difference between pressurizer and main spray

N_A = Allowable number of spray cycles

N = Number of cycles in ΔT range indicated

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager-Nuclear shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President - Nuclear Operations, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room.

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed or performed by a licensed Operator or licensed Senior Operator and supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least five members shall be maintained on site at all times.* The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

*The Health Physics Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of individuals of the nuclear power plant operating staff who are responsible for manipulating plant controls or for adjusting on-line systems and equipment affecting plant safety which would have an immediate impact on public health and safety.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, the following guidelines shall be followed:

1. An individual shall not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual shall not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3. A break of at least 8 hours shall be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime shall be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager, the assistant Plant Managers, the Operations Superintendent - Nuclear or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime will be reviewed monthly by the Station Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

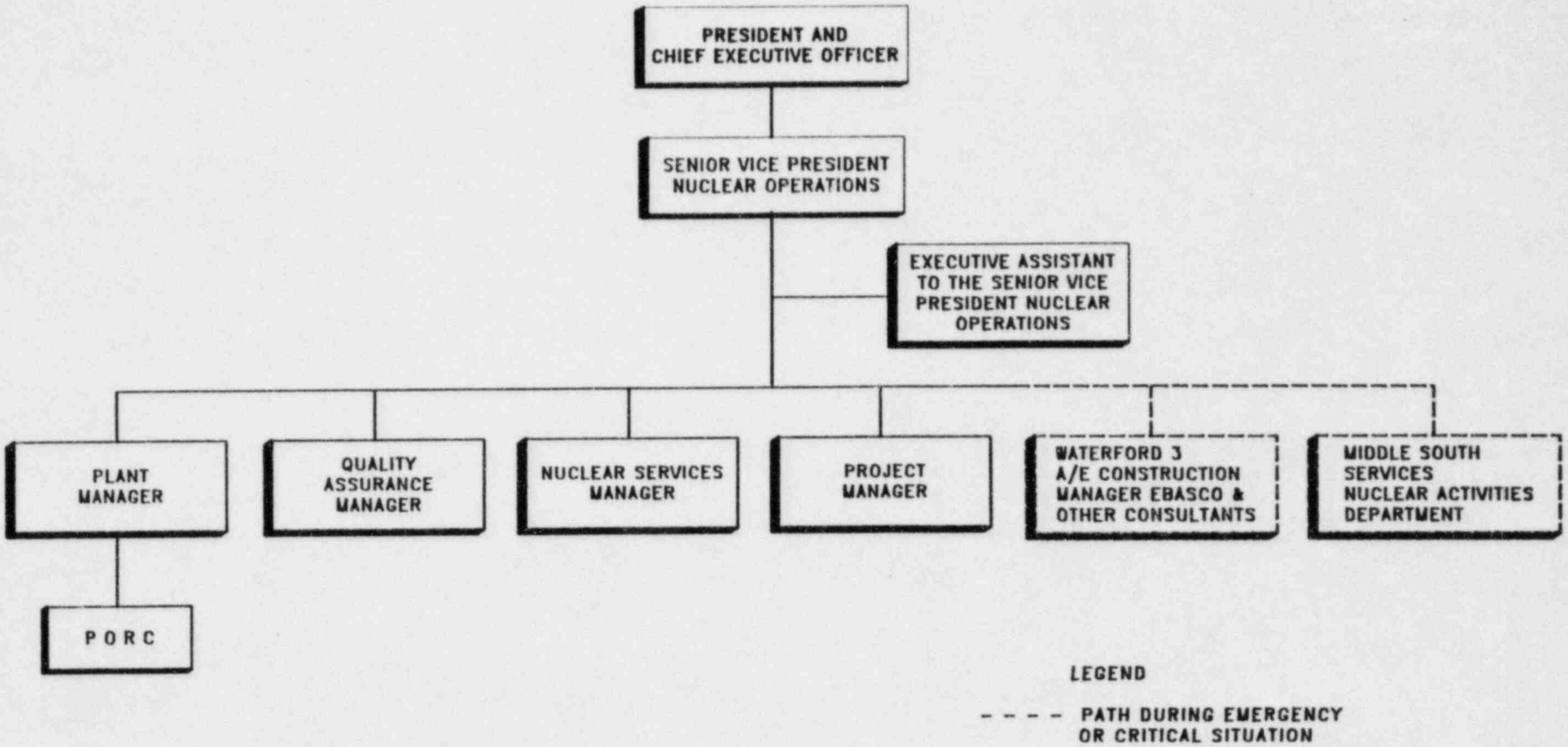

FIGURE 6.2-1

OFFSITE ORGANIZATION FOR MANAGEMENT AND TECHNICAL SUPPORT

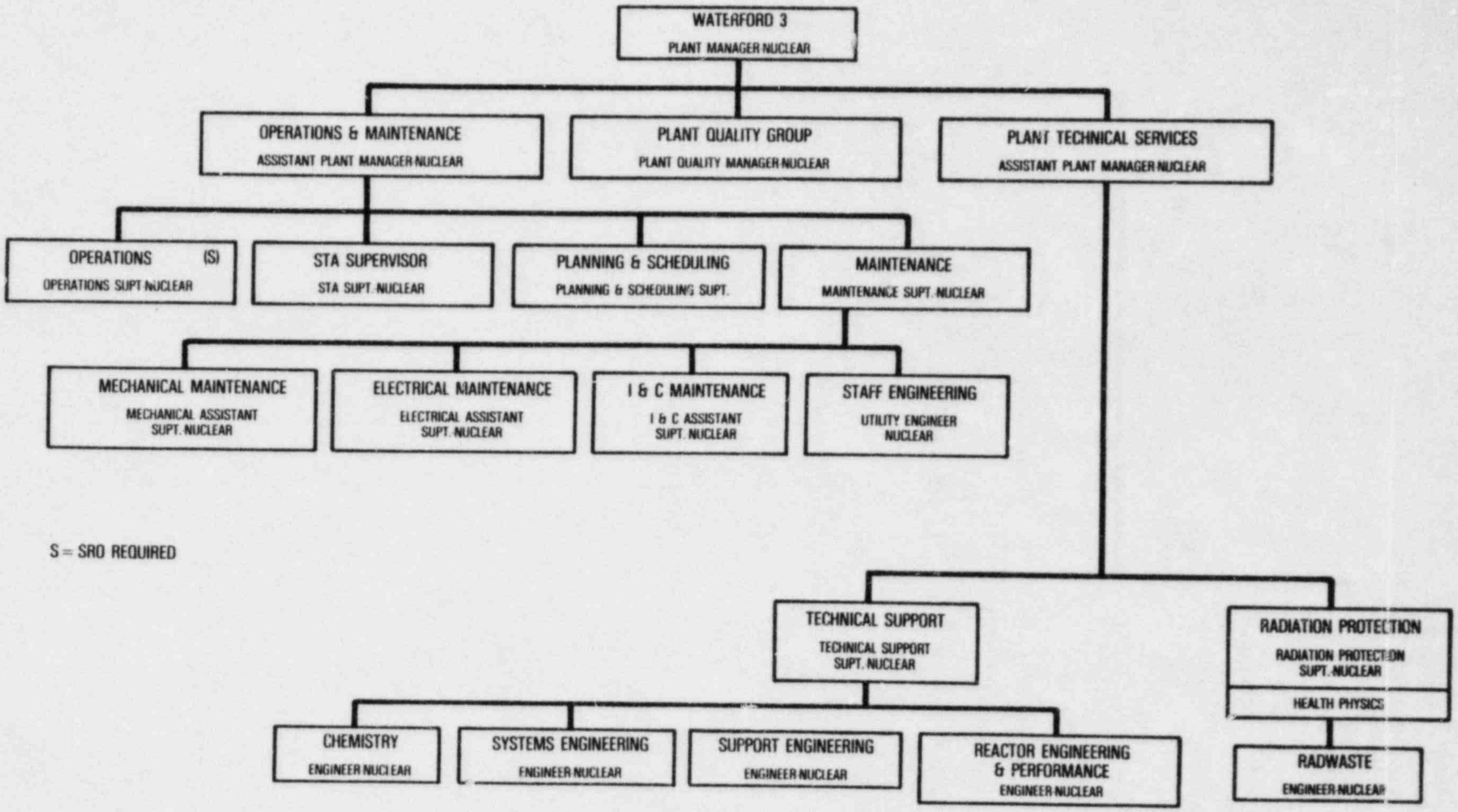

FIGURE 6.2-2

PLANT OPERATIONS ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

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

Except for the Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2-1 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 MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license 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 unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Engineering and Nuclear Safety Manager.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit 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 is an onsite independent technical review group that reports offsite to the Engineering and Nuclear Safety Manager. The ISEG shall have the authority necessary to perform the functions and responsibilities as delineated above.

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Engineering and Nuclear Safety Manager.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

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*Not responsible for sign-off function.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 except that:

a. The Radiation Protection Superintendent shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

b. Personnel in the Health Physics, Chemistry and Radwaste Departments shall meet or exceed the minimum qualifications of ANSI N18.1-1971.

c. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

d. Personnel in the Plant Quality Department, and other staff personnel who peform inspection, examination, and testing functions, shall meet or exceed the minimum qualifications of Regulatory Guide 1.58, Rev. 1, September 1980. (Endorses ANSI N45.2.6-1978)

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager-Nuclear and shall meet or exceed the requirements and recommendations of Section 5.2 of ANSI 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager-Nuclear on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Assistant Plant Manager-Nuclear (Plant Technical Services or Operations and Maintenance)
Vice Chairman:	Technical Support Superintendent-Nuclear
Member:	Maintenance Superintendent-Nuclear
Member:	Operations Superintendent-Nuclear
Member:	Radiation Protection Superintendent-Nuclear
Member:	Quality Control Manager-Nuclear

ALTERNATES

6.5.1.3 The PORC is chaired by one of the Assistant Plant Managers and, in their absence, the Technical Support Superintendent acts as Chairman. If all three are absent, the Plant Manager-Nuclear will appoint a temporary Chairman. All other alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and three members, including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for the below listed activities. The PORC may delegate the performance of reviews but will maintain cognizance over and responsibility for them.

a. Review of (1) all procedures required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager-Nuclear to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

e. Review of investigations of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Plant Manager-Nuclear and to the Safety Review Committee.

f. Review of all REPORTABLE EVENTS.

RESPONSIBILITIES (Continued)

g. Review of unit operations to detect potential hazards to nuclear safety.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager-Nuclear or the Safety Review Committee.

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Safety Review Committee.

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Safety Review Committee.

k. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PORC meeting.

l. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

m. Review of any accidental, unplanned or uncontrolled radioactive release including reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President, Nuclear Operations and to the Safety Review Committee.

n. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL, and major changes to radwaste treatment systems.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager-Nuclear, prior to implementation except as provided in Specification 6.8.3, approval or disapproval of items considered under Specification 6.5.1.6a. through d. and l.

b. Render determinations in writing, prior to implementation except as provided in Specification 6.8.3, with regard to whether or not each item considered under Specification 6.5.1.6a. through e. constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Senior Vice President-Nuclear Operations and the Safety Review Committee of disagreements between the PORC and the Plant Manager-Nuclear; however, the Plant Manager-Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Senior Vice President-Nuclear Operations and the Safety Review Committee.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering and

h. Quality assurance practices.

COMPOSITION

6.5.2.2 The SRC shall be composed of at least five members, including the Chairman. Members of the SRC may be from within the LP&L organization or from organizations external to LP&L.

The qualifications of members selected for the SRC shall be in accordance with Section 4.7 of ANSI/ANS 3.1-1978.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the SRC Chairman to provide expert advice to the SRC.

MEETING FREQUENCY

6.5.2.5 The SRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the SRC necessary for the performance of the review and audit function of these technical specifications shall consist of a minimum of five members or of not less than a majority of the composition of members in Specification 6.5.2.2, whichever is greater. No more than a minority of the members shall have line responsibility for operation of the plant.

REVIEW

6.5.2.7 The SRC shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PORC.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the SRC. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the SRC or the Vice President-Nuclear Operations.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The Primary Coolant Sources Outside Containment Program at least once per 24 months.

i. The In-Plant Radiation Monitoring Program at least once per 24 months.

j. The Secondary Water Chemistry Program at least once per 24 months.

k. The Post-Accident Sampling Program at least once per 24 months.

l. The Basemat Monitoring Program at least once per 24 months.

m. The radiological environmental monitoring program and the results thereof at least once per 12 months.

n. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

o. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

AUDITS (Continued)

p. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.2.9 The SRC shall report to and advise the Senior Vice President- Nuclear Operations on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Senior Vice President-Nuclear Operations within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Senior Vice President-Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of this review shall be submitted to the SRC and the Senior Vice President-Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

SAFETY LIMIT VIOLATION (Continued)

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President-Nuclear Operations and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Senior Vice President-Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants, including independent verification of modified constants.

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PORC.

h. Administrative procedures implementing the overtime guidelines of Specification 6.2.2f., including provisions for documentation of deviations.

i. PROCESS CONTROL PROGRAM implementation.

PROCEDURES AND PROGRAMS (Continued)

j. OFFSITE DOSE CALCULATION MANUAL implementation.

k. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5.1.6 and shall be approved by the Plant Manager-Nuclear prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected;

c. The change is documented, reviewed by the PORC and approved by the Plant Manager-Nuclear within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the containment spray, safety injection, hydrogen analyzer, and the post-accident sampling system. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

PROCEDURES AND PROGRAMS (Continued)

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

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical variables and control points for these variables,
2. Identification of the procedures used to measure the values of the critical variables,
3. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,
4. Procedures for the recording and management of data,
5. Procedures defining corrective actions for all off-control point chemistry conditions, and
6. A procedure 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.

d. Post-accident 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:

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

e. Basemat Monitoring*

A program for monitoring of the Nuclear Plant Island Structure (NPIS) Common Foundation Basemat to ensure the continued integrity of the Basemat. The program shall include:

1. settlement of the basemat
2. changes in ground water chemistry that could effect corrosion of reinforcing steel
3. seasonal variation in ground water levels
4. mapping of significant cracking in the basemat and adjacent walls.

*Not required until prior to exceeding 5% of RATED THERMAL POWER.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

*This tabulation supplements the requirements of K20.407 of 10 CFR Part 20.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

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*One map shall cover stations near the SITE BOUNDARY a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report shall also include once a year an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

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*In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

INDUSTRIAL SURVEY OF TOXIC OR HAZARDOUS CHEMICALS REPORT

6.9.1.9 Surveys and analyses of major industries in the vicinity of Waterford 3 which could have significant inventories of toxic chemicals onsite to determine impact on safety shall be performed and submitted to the Commission at least once every 4 years.

6.9.1.10 A survey of major pipelines ($\geq$ 4 inches) within a 2-mile radius of Waterford 3, which contain explosive or flammable materials and may represent a hazard to Waterford 3, including scaled engineering drawings or maps which indicate the pipeline locations, shall be performed and submitted to the Commission at least once every 4 years.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 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.

RECORD RETENTION (Continued)

6.10.2 The following records shall be retained for at least 5 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.

c. All REPORTABLE EVENTS.

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.

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.

6.10.3 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 inservice inspections performed pursuant to these Technical Specifications.

RECORD RETENTION (Continued)

i. Records of quality assurance activities required by the QA 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 and the SRC.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of audits performed under the requirements of Specification 6.5.2.8.

o. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 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.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/h but less than 1000 mrem/h shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

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*Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Radiation Protection Superintendent-Nuclear in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrems* shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems* that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (Continued)

2. A determination that the change did not reduce the overall conformance of the waste product to existing criteria of 10 CFR Part 61 and for burial at low-level waste sites; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3. Documenation of the fact that the change has been reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the PORC.

b. Shall become effective upon review and acceptance by the PORC.

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16. ABSTRACT *(200 words or less)*

The Waterford, Unit 3 Technical Specifications were prepared by the U. S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
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APPENDIX B

TO FACILITY OPERATING LICENSE NO. DPR-26

WATERFORD STEAM ELECTRIC STATION - UNIT NO. 3

LOUISIANA POWER & LIGHT COMPANY

DOCKET NO. 50-382

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

DECEMBER 1984

WATERFORD STEAM ELECTRIC STATION - UNIT NO. 3

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during construction and operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the plant is operated in a environmentally acceptable manner, as established by the FES-OL and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State, and local requirements for environmetnal protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters and regulated by way of the licensee's NPDES permit.

2.0 Environmental Protection Issues

In the FES-OL dated September 1981, the staff considered the environmental impacts associated with the operation of Waterford Steam Electric Station Unit No. 3. The environmental resources which were evaluated, and the corresponding NRC staff decisions regarding the protection of each resource are as follows:

2.1 Aquatic Resources Issues

Effluent limitations and monitoring requirements are contained in the NPDES Permit (No. LA 0007374) issued by the U.S. Environmental Protection Agency, Region VI. The NRC will rely on EPA for regulation of matters involving the protection of water quality and aquatic biota.

2.2 Terrestrial Resources Issues

No terrestrial resources issues were raised by the staff in the FES-OL.

2.3 Cultural Resources Issues

There are two archeological sites on the licensee's property which have been determined to be eligible for the National Register of Historic Places. Protection for these two sites will be provided through a Cultural Resources Protection Plan. NRC requirements with regard to the cultural resources issue are specified in Subsection 4.2.1 of this EPP.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment, provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP. Changes in plant design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in construction or operation activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological effects are confined to the on-site areas previously distrubed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level [in accordance with 10 CFR Part 51.5(b)(2)] or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of its Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the NPDES Permit and State Certification

Violations of the NPDES Permit or the State certification (pursuant to Section 401 of the Clean Water Act) shall be reported to the NRC by providing copies of the reports required by the NPDES Permit or State Certification. The licensee shall also provide the NRC with copies of the results of studies, other than routine monitoring, conducted in accordance with the NPDES Permit at the same time they are submitted to the permitting agency.

Changes to, or renewals of, the NPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurance of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours, followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Cultural Resources Protection Plan

The licensee, the NRC and the State Historic Preservation Officer (SHPO) concur that the appropriate action to be taken to negate any possible adverse effects to the Waterford 3 cultural resources by the operation and maintenance activities of the licensee will be through a Cultural Resources Protection Plan that provides documentation of a "no adverse effect" determination.

This Cultural Resources Protection Plan was transmitted to the NRC and the SHPO by a letter from L. V. Maurin to G. W. Knighton, dated April 15, 1983 for final review and concurrence, after which the NRC submitted the plan to the Advisory Council on Historic Preservation (ACHP) for comment on September 28, 1983. ACHP concurrence was received by the NRC, on October 18, 1983 without change.

The Cultural Resources Protection Plan, as referenced above, is the binding document to which the licensee will adhere and this Section of the EPP is considered fully satisfied with no further action required.

5.0 Adminstrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organizational structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner covenient for review and inspection. These records and logs shall be made available to the NRC on request.

Records of modifications to plant structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the plant. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Request for a change in the Environmental Protection plan shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the Environmental Protection Plan.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May of the year following issuance of the operating license. The period of the first report shall begin with the date of issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this Environmental Protection Plan for the report period, including a comparison with any related preoperational studies, operational controls (as appropriate), and previous non-radiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environmental are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating actions.

The Annual Environmental Operating Report shall also include:

(a) A list of EPP noncompliances and the corrective actions taken to remedy them.

(b) A list of all changes in station design or operation, tests, experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(c) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of and event described in Section 4.1 of this Plan. The reports shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

DEC 18 1984

APPENDIX C

ANTITRUST CONDITIONS
FACILITY OPERATING LICENSE NO. NPF-26

(1) (a) As used herein, "entity" shall mean any municipality, rural electric cooperative, public or private corporation, governmental agency such as TVA and Southwest Power Administration, or lawful association of any of the foregoing (a) which lawfully exists and owns and operates or proposes in good faith to own or operate facilities for generation of electric power and energy; (b) which, with exception of municipalities, rural electric cooperatives and governmental agencies, is or will upon commencement of operations be a public utility (or in the case of an association each member thereof, excepting municipalities, rural electric cooperatives and governmental agencies, is a public utility) under the law of Louisiana and the Federal Power Act and provides or upon commencement of operations will provide electric service under contracts or rate schedules on file with and subject to regulation of the Louisiana Public Service Commission and the FPC; and (c) with which applicant has or may feasibly have a physical interconnection within the State of Louisiana.

For the purposes of paragraphs 5 and 6 hereof, any person who would otherwise qualify as an "entity" herein above except for not meeting the requirements of 1(a) shall be considered an "entity" if that person owns or operates or proposes in good faith to own or operate facilities for generation, transmission and/or distribution of electric power and energy.

(b) "Cost" means any operating and maintenance expenses involved together with any ownership costs which are reasonably allocable to the transaction consistent with power pooling practices (where applicable). No value shall be included for loss of revenues from sale of power at wholesale or retail by one party to a customer which another party might otherwise serve. Cost shall include a reasonable return on the applicant's investment. The sale of a portion of the capacity of a generating unit shall be upon the basis of a rate that will recover to the seller the pro rata part of the fixed costs and operating and maintenance expenses of the unit, provided that, in circumstances in which the applicant and one or more entities in Louisiana take an undivided interest in a unit in fee, construction costs and operation and maintenance expenses shall be paid pro rata.

(2) (a) The applicant shall interconnect and share reserves on an equalized percentage reserve basis with any entity in Louisiana which engages in or proposes to engage in electric generation and/or bulk power purchases on terms that will provide for the applicant's costs, and allow the other participant(s) full access to the benefits of reserve sharing coordination, and in addition, shall include but not be limited to emergency service, scheduled maintenance service, and establishing reserves. Such interconnection shall be at a voltage and capacity requested by such entity whenever it is economically feasible for the parties.

(b) Emergency service and/or scheduled maintenance service to be provided by each party shall be furnished to the fullest extent available from the supplying party and desired by the party in need. The applicant and each party(ies) shall provide to the other emergency service and/or scheduled maintenance service if and when available from its own generation and from generation of others to the extent it can do so without impairing service to its customers including other electric systems to whom it has firm commitments.

(c) The applicant and the other party(ies) to a reserve sharing arrangement shall from time to time jointly establish the minimum reserves to be installed and/or provided under contractual arrangements as necessary to maintain in total a reserve margin sufficient to provide adequate reliability of power supply to the interconnected systems of the parties. If the applicant plans its reserve margin on a pooled basis with other Middle South System companies, the reserves jointly established hereunder shall be on the same basis. Unless otherwise agreed upon, minimum reserves shall be calculated as a percentage of estimated peak load responsibility. No party to the arrangement shall be required to maintain greater reserves than the percentage of its estimated peak load responsibility which results from the aforesaid calculation, provided that, if the reserve requirements of the applicant are increased over the amount the applicant would be required to maintain without such interconnection, then the other party(ies) shall be required to carry or provide for as its (their) reserves the full amount in kilowatts of such increase.

(d) The parties to such a reserve sharing arrangement shall provide such amounts of ready reserve capacity as may be adequate to avoid the imposition of unreasonable demands on the other in meeting the normal contingencies of operating its system. However, in no circumstances shall the ready reserve requirement exceed the installed reserve requirement.

(e) Interconnections will not be limited to low voltages when higher voltages are available from the applicant's installed facilities in the area where interconnection is desired, when the proposed arrangement is found to be technically and economically feasible. Control and telemetering facilities shall be provided as required for safety and prudent operation of the interconnected systems.

(f) Interconnection and coordination agreements shall not embody any restrictive provisions pertaining to intersystem coordination. Good industry practice as developed in the area from time to time (if non-restrictive) will satisfy this provision.

(3) The applicant will purchase (when needed) or sell (when available) "unit power" or "deficiency power" at mutually agreed upon delivery points on or adjacent to its transmission system from or to any entity engaging in or proposing to engage in electric generation and/or bulk power purchases at the cost (including a reasonable return) of new power supply, as distinguished from average system cost, when such transaction would serve to reduce the overall cost of new bulk power supply for itself and the other participant to the transaction.

(4) With respect to Waterford Unit No. 3 and any future nuclear generating plant or unit of the applicant, or any plant or unit in which the applicant may acquire an interest in Louisiana, any entity that expresses an interest in participation will be offered (1) for Waterford Unit No. 3 and for any future nuclear generating plant or unit of the applicant, the opportunity to have access* to a portion of the plant or unit capacity, or (2) with respect to any plant or unit in which the applicant may acquire an interest, the opportunity to have access* to a portion of the plant or unit capacity to the extent the applicant is able; in either event, upon the basis of a rate that will recover to the applicant the average fixed costs (including a reasonable return) of the

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*"The opportunity to have access" shall be for a period of one year after the applicant has provided to each enquiring entity financial data, which in the opinion of the Regulatory staff of the Commission is sufficient to enable such entity to make a feasibility study as to participation. The applicant shall provide such financial data as soon as reasonably feasible after receiving an inquiry. As to any entity or some or all entities in Louisiana the applicant can start the running of the aforesaid one year period by supplying to it or them, without waiting for an inquiry, the aforesaid financial data.

plant or unit or the applicant's interest in any plant or unit.** The entity receiving such power will pay the associated energy, maintenance, and operating costs incurred for the power it receives. In connection with this access, the applicant will also offer transmission service to the geographic extent of its then existing transmission system for delivery of such power to such purchasing entity on a basis that will fully compensate the applicant for its transmission costs (including a reasonable return).

In the event that the law of Louisiana should be changed to the extent that property owned jointly is not susceptible to partition and that such joint ownership is not otherwise an impediment to financing, the Company must, in accordance with the provisions of its Commitment 4, offer joint ownership in any future nuclear generating plant or unit owned by it (or in which it may acquire an interest in Louisiana) to any entity requesting such access.

In the event that during the term of the instant license, or any extension or renewal thereof, the applicant participates in the ownership of or obtains rights to, and obligations in, a portion of the output of one or more nuclear generating units constructed, owned or operated by an affiliate or subsidiary of the Middle South Utilities System other than the applicant or by any successor in title to the Waterford Nuclear Unit, the applicant shall exert its best efforts to obtain participation in such nuclear unit by an entity(ies) in the State of Louisiana requesting such participation on terms equivalent to the terms of the applicant's participation therein. In connection with such participation, the applicant will also offer transmission service to the geographic extent of its then existing transmission system for delivery of such power to such purchasing entity on a basis that will fully compensate the applicant for its transmission cost (including a reasonable return).

For the purposes of this paragraph, any person who would otherwise qualify as an "entity" except for the lack of a physical interconnection with the applicant shall be considered an "entity" if that person is or will be interconnected with an "entity" or member of the Southwest Power Pool which is interconnected with the applicant.

**Nothing herein shall be deemed to exclude the participation of an entity through a prepaid unit power basis should such participation be economically, technically and legally feasible. Moreover, nothing herein shall be deemed to exclude participation of an entity on a joint venture basis in Waterford Unit 3 if the Company shall in its sole discretion decide to enter into such a joint venture.

(5) The applicant shall transmit power and energy over its transmission facilities among entities in the State of Louisiana with which it is interconnected and has or will have a transmission schedule in effect. For each coordinating group of entities there shall be a single transmission charge. In addition, for any entity with whom applicant is interconnected, the applicant will transmit to or from that entity's then existing interconnection with the applicant, power delivered to the applicant by another entity (or from the applicant to another entity) whose transmission facilities adjoin those of the applicant, provided (1) there is or will be a transmission schedule in effect, and (2) the arrangements reasonably can be accommodated from a functional and technical standpoint. The transmission of such power and energy shall be at a rate that will fully compensate the applicant for its costs (including a reasonable return) for the use of its system. Any entity or group of entities requesting such transmission arrangements shall give reasonable advance notice of its schedule and requirements. (The foregoing applies to any entities to which the applicant may be interconnected in the future as well as those to which it is now interconnected.)

The applicant shall include in its planning and construction program sufficient transmission capacity as required for the transactions referred to in the above paragraph, and in those instances where such transactions are consummated, a transmission schedule(s) shall be placed in effect; provided that any entity in the State of Louisiana give the applicant sufficient advance notice as may be necessary to accommodate its requirements from a functional and technical standpoint and that such entity fully compensates the applicant for its cost (including a reasonable return). The applicant shall not be required to construct transmission facilities which will be of no demonstrable present or future benefit to the applicant.

For the purposes of this paragraph, (1) any person in the State of Louisiana who would otherwise qualify as an "entity" except for the lack of a physical interconnection with the applicant shall be considered an "entity" if that person is or will be interconnected with an "entity" or member of the Southwest Power Pool which is interconnected with the applicant; and (2) Arkansas Power and Light Company, Mississippi Power and Light Company, and Mississippi Power Company, or any successor thereof, shall also be considered "entities."

(6) The applicant will enter into arrangements mutually agreed upon for the sale of power and energy under its effective [rate schedule] tariffs to any entity that owns an electric distribution system and has or may feasibly have a physical interconnection within the State of Louisiana. In connection with such arrangements, the applicant shall not be required to construct facilities which will be of no demonstrable present or future benefit to the applicant.

(7) It is recognized that the foregoing conditions are to be implemented in a manner consistent with the provisions of the Federal Power Act to the extent applicable, and all rates, charges or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.