UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-317

CALVERT CLIFFS NUCLEAR POWER PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
License No. DPR-53

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated September 17, 1993, as supplemented on January 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-53 is hereby amended to read as follows:

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BALTIMORE GAS AND ELECTRIC COMPANY

DOCKET NO. 50-318

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 163
License No. DPR-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Baltimore Gas and Electric Company (the licensee) dated September 17, 1993, as supplemented on January 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-69 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 163 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 17, 1994

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 186 FACILITY OPERATING LICENSE NO. DPR-53

AMENDMENT NO. 163 FACILITY OPERATING LICENSE NO. DPR-69

DOCKET NOS. 50-317 AND 50-318

Revise Appendix A as follows: DPR-53*

Remove Pages	Insert Pages
I-III	I-III
VI-VII	VI-VII
XII	XII
XVII-XVIII	XVII-XVIII
1-1	1-1
1-3 thru 1-8	1-3 thru 1-8
2-1 thru 2-13	2-1 thru 2-6
B2-1	B2-1
B2-3 thru B2-6	B2-3 thru B2-6
3/4 1-1 thru 3/4 1-36	3/4 1-1 thru 3/4 1-33
3/4 2-1 thru 3/4 2-17	3/4 2-1 thru 3/4 2-8
3/4 3-27 thru 3/4 3-29	3/4 3-27 thru 3/4 3-29
3/4 7-1	3/4 7-1
3/4 7-3 thru 3/4 7-48	3/4 7-3 thru 3/4 7-47
3/4 9-1	3/4 9-1
B3/4 1-1 thru B3/4 1-5	B3/4 1-1 thru B3/4 1-5
B3/4 2-1 thru B3/4 2-2	B3/4 2-1 thru B3/4 2-2
B3/4 7-1 thru B3/4 7-8	B3/4 7-1 thru B3/4 7-7
B3/4 9-1	B3/4 9-1
6-22 thru 6-27	6-22 thru 6-31

*Rollover pages - vertical line at page number only.

2-3
3/4 1-4 thru 3/4 1-5
3/4 1-8
3/4 1-10
3/4 1-12 thru 3/4 1-13
3/4 1-17 thru 3/4 1-20
3/4 1-22
3/4 1-26
3/4 1-29
3/4 1-33
3/4 7-4 thru 3/4 7-47
B3/4 1-2
B3/4 7-2 thru B3/4 7-7
6-27 thru 6-31

Revise Appendix A as follows: DPR-69*

Remove Pages	Insert Pages
I-III	I-III
VI-VII	VI-VII
XII	XII
XVII-XVIII	XII-XVIII
1-3 thru 1-8	1-3 thru 1-8
2-1 thru 2-13	2-1 thru 2-6
B2-1	B2-1
B2-3 thru B2-7	B2-3 thru B2-7
3/4 1-1 thru 3/4 1-36	3/4 1-1 thru 3/4 1-33
3/4 2-1 thru 3/4 2-17	3/4 2-1 thru 3/4 2-8
3/4 3-28 thru 3/4 3-29	3/4 3-28 thru 3/4 3-29
3/4 7-1	3/4 7-1
3/4 7-3 thru 3/4 7-48	3/4 7-3 thru 3/4 7-47
3/4 9-1	3/4 9-1
B3/4 1-1	B3/4 1-1
B3/4 1-3	B3/4 1-3
B3/4 1-5	B3/4 1-5
B3/4 2-1 thru B3/4 2-2	B3/4 2-1 thru B3/4 2-2
B3/4 7-1 thru B3/4 7-8	B3/4 7-1 thru B3/4 7-7
B3/4 9-1	B3/4 9-1
6-22 thru 6-27	6-22 thru 6-31

*Rollover pages - vertical line at page number only.

2-3
B2-7
3/4 1-4 thru 3/4 1-5
3/4 1-8
3/4 1-10
3/4 1-12 thru 3/4 1-13
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3/4 1-17 thru 3/4 1-20
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1.0 DEFINITIONS

DEFINED TERMS

1.1 The **DEFINED TERMS** of this section appear in capitalized type, bold, and are applicable throughout these Technical Specifications.

ACTION

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

AXIAL SHAPE INDEX

1.3 The **AXIAL SHAPE INDEX** (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The **AXIAL SHAPE INDEX** (Y_I) used for the trip and pretrip signals in the Reactor Protection System is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

AZIMUTHAL POWER TILT - T_q

1.4 **AZIMUTHAL POWER TILT** shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

CHANNEL CALIBRATION

1.5 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.

CONTROLLED LEAKAGE

1.9 **CONTROLLED LEAKAGE** shall be the water flow from the reactor coolant pump seals.

CORE ALTERATION

1.10 **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.

CORE OPERATING LIMITS REPORT

1.11 The **CORE OPERATING LIMITS REPORT** is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

1.12 **DOSE EQUIVALENT I-131** shall be that concentration of I-131 (μCi/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

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A **GASEOUS RADWASTE TREATMENT SYSTEM** is any system designed and installed to reduce radioactive gaseous effluent by collecting Primary Coolant System offgases from the Primary System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 **IDENTIFIED LEAKAGE** shall be:

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

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

c. Reactor Coolant System leakage through a steam generator to the Secondary System.

MEMBER(S) OF THE PUBLIC

1.18 **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 utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.19 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 effluent, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.20 A system, subsystem, train, component or device shall be **OPERABLE** or have **OPERABILITY** when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other required 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

1.21 An **OPERATIONAL MODE** shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.22 **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 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.23 **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)

1.24 The **PROCESS CONTROL PROGRAM** shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State and local regulations governing the disposal of the radioactive waste.

PURGE - PURGING

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

RATED THERMAL POWER

1.26 **RATED THERMAL POWER** shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 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

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

SHUTDOWN MARGIN

1.29 **SHUTDOWN MARGIN** shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length 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

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

SOLIDIFICATION

1.31 **SOLIDIFICATION** shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

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.

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

1.35 The **TOTAL INTEGRATED RADIAL PEAKING FACTOR** is the ratio of the peak pin power to the average pin power in an unrodded core.

TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

1.36 The **TOTAL PLANAR RADIAL PEAKING FACTOR** is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes.

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.38 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

1.39 A **VENTILATION EXHAUST TREATMENT SYSTEM** is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluent 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 effluent. Engineered Safety Feature (ESF) Atmospheric Cleanup Systems are not considered to be **VENTILATION EXHAUST TREATMENT SYSTEM** components.

VENTING

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of **THERMAL POWER**, pressurizer pressure, and highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1-1.

APPLICABILITY: **MODES** 1 and 2.

ACTION: Whenever the point defined by the combination of the highest operating loop cold leg temperature and **THERMAL POWER** has exceeded the appropriate pressurizer pressure line, be in **HOT STANDBY** within 1 hour.

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.

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.

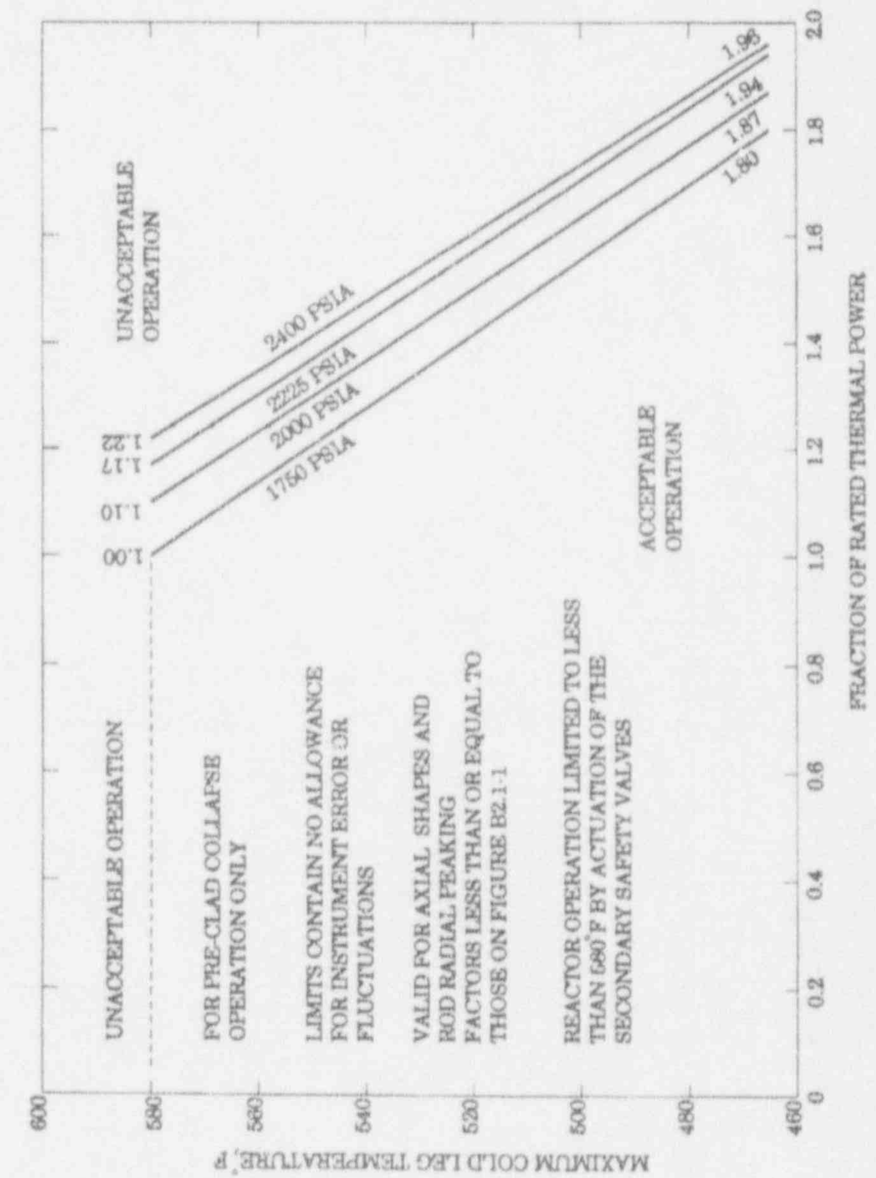

FIGURE 2.1-1

REACTOR CORE THERMAL MARGIN SAFETY LIMIT

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 3.3.1.1 until the channel is restored to **OPERABLE** status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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. Power Level - High	$\leq$ 10% above THERMAL POWER, with a minimum setpoint of 30% of RATED THERMAL POWER, and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.	$\leq$ 10% above THERMAL POWER, and a minimum setpoint of 30% of RATED THERMAL POWER and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)	$\geq$ 95% of design reactor coolant flow*	$\geq$ 95% of design reactor coolant flow*
4. Pressurizer Pressure - High	$\leq$ 2400 psia	$\leq$ 2400 psia
5. Containment Pressure - High	$\leq$ 4 psig	$\leq$ 4 psig
6. Steam Generator Pressure - Low (2)	$\geq$ 685 psia	$\geq$ 685 psia
7. Steam Generator Water Level - Low	$\geq$ 10 inches below top of feed ring	$\geq$ 10 inches below top of feed ring

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Axial flux offset (3)	Trip setpoint adjusted to not exceed the limits provided in the COLR	Trip setpoint adjusted to not exceed the limits provided in the COLR
9. Thermal Margin/Low Pressure (1)		
a. Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limits provided in the COLR	Trip setpoint adjusted to be not less than the larger of (1) 1875 psia, or (2) the limits provided in the COLR
b. Steam Generator Pressure Difference - High (1)	$\leq$ 135 psid	$\leq$ 135 psid
10. Loss of Load	NA	NA
11. Rate of Change of Power - High (4)	$\leq$ 2.6 decades per minute	$\leq$ 2.6 decades per minute

TABLE 2.2-1 (Continued)

TABLE NOTATION

* See Specification 3.2.5, "DNB Parameters," for the design reactor coolant flow.

(1) Trip may be bypassed below 10^{-4}% OF **RATED THERMAL POWER**; bypass shall be automatically removed when **THERMAL POWER** is ≥ 10^{-4}% of **RATED THERMAL POWER**.

(2) Trip may be manually bypassed below 785 psia; bypass shall be automatically removed at or above 785 psia.

(3) Trip may be bypassed below 15% of **RATED THERMAL POWER**; bypass shall be automatically removed when **THERMAL POWER** is ≥ 15% of **RATED THERMAL POWER**.

(4) Trip may be bypassed below 10^{-4}% and above 12% of **RATED THERMAL POWER**.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible ladding perforation which could result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 22.0 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by 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.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and, therefore, **THERMAL POWER** and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the DNB SAFDL of 1.15 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This DNB SAFDL assures with at least a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curves of Figure 2.1-1 show conservative loci for points of **THERMAL POWER**, Reactor Coolant System pressure and maximum cold leg temperature for which the DNB SAFDL is not violated for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. The area of safe operation is below and to the left of these lines. Reactor operation at **THERMAL POWER** levels higher than 110% of **RATED THERMAL POWER** is prohibited by the high power level trip setpoint specified in Table 2.2-1.

2.1 SAFETY LIMITS

BASES

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Reactor Protective System in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and **THERMAL POWER** level that would result in a DNBR of less than 1.15, in conjunction with the ESCU methodology, and preclude the existence of flow instabilities.

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 pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I, 1968 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component 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 parameter. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their safety limits. 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 the trip setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

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

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated **THERMAL POWER** level. Operator action is required to increase the trip setpoint as **THERMAL POWER** is increased. The trip setpoint is automatically decreased as **THERMAL POWER** decreases. The trip setpoint has a maximum value of 107.0% of **RATED THERMAL POWER** and a minimum setpoint of 30% of **RATED THERMAL POWER**. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state **THERMAL POWER** level at which a trip would be actuated is 110% of **RATED THERMAL POWER**, which is the value used in the safety analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. The low-flow trip setpoints and Allowable Values have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above the DNB SAFDL of 1.15, in conjunction with the ESCU methodology, under normal operation and expected transients.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection.

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 setting of 685 psia is sufficiently below the full-load operating point of 850 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 setting was used with an uncertainty factor of ± 85 psi which was based on the main steam line break event inside containment.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the Reactor Coolant System will not exceed its Safety Limit. The specified setpoint in combination with the Auxiliary Feedwater Actuation System ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore to incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power and reactor inlet temperature. The minimum value of reactor coolant flow rate, the maximum **AZIMUTHAL POWER TILT** and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints include allowances for equipment response time, measurement uncertainties, processing error and a further allowance of 40 psia to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The **SHUTDOWN MARGIN** shall be within the limit provided in the COLR.

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

ACTION: With the **SHUTDOWN MARGIN** outside the limit provided in the COLR, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or equivalent until the required **SHUTDOWN MARGIN** is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The **SHUTDOWN MARGIN** shall be determined to be within the limit provided in the COLR.

a. Within one 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 untrippable, the above required **SHUTDOWN MARGIN** shall be increased by an amount at least equal to the withdrawn worth of the untrippable CEA(s).

b. When in **MODES** 1 or 2 with K_{eff} ≥ 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.

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* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

c. When in **MODE** 2 with $K_{eff} < 1.0$, 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.

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 **MODES** 3 or 4, 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.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 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 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The **SHUTDOWN MARGIN** shall be within the limit provided in the COLR, and when pressurizer level is less than 90 inches from the bottom of the pressurizer, all sources of non-borated water shall be ≤ 88 gpm.

APPLICABILITY: **MODE** 5.

ACTION:

a. With the **SHUTDOWN MARGIN** outside the limit provided in the COLR, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or equivalent until the required **SHUTDOWN MARGIN** is restored.

b. With the pressurizer drained to < 90 inches and all sources of non-borated water > 88 gpm, immediately suspend all operations involving positive reactivity changes while the **SHUTDOWN MARGIN** is increased to compensate for the additional sources of non-borated water or reduce the sources of non-borated water to ≤ 88 gpm.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The **SHUTDOWN MARGIN** shall be determined to within the limit provided in the COLR:

a. Within one 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 untrippable, the above required **SHUTDOWN MARGIN** shall be increased by an amount at least equal to the withdrawn worth of the 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,

SURVEILLANCE REQUIREMENTS (Continued)

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

4.1.1.2.2 With the pressurizer drained to < 90 inches determine:

a. Within one hour and every 12 hours thereafter that the level in the Reactor Coolant System is above the bottom of the hot leg nozzles, and

b. Within one hour and every 12 hours thereafter that the sources of non-borated water are $\leq$ 88 gpm or the **SHUTDOWN MARGIN** has been increased to compensate for the additional non-borated water sources.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.1 BORATION CONTROL

Boron Dilution

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the Reactor Coolant System shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: All **MODES**.

ACTION: With the flow rate of reactor coolant through the Reactor Coolant System < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the Reactor Coolant System shall be determined to be $\geq$ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying $\geq$ 3000 gpm through the Reactor Coolant System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

Moderator Temperature Coefficient

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the limit provided in the COLR, and

b. Less positive than the limit line of Figure 3.1.1-1.

APPLICABILITY: **MODES** 1 and 2 with $K_{eff} \geq 1.0^{\#}$.

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

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 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.

\# See Special Test Exception 3.10.2.

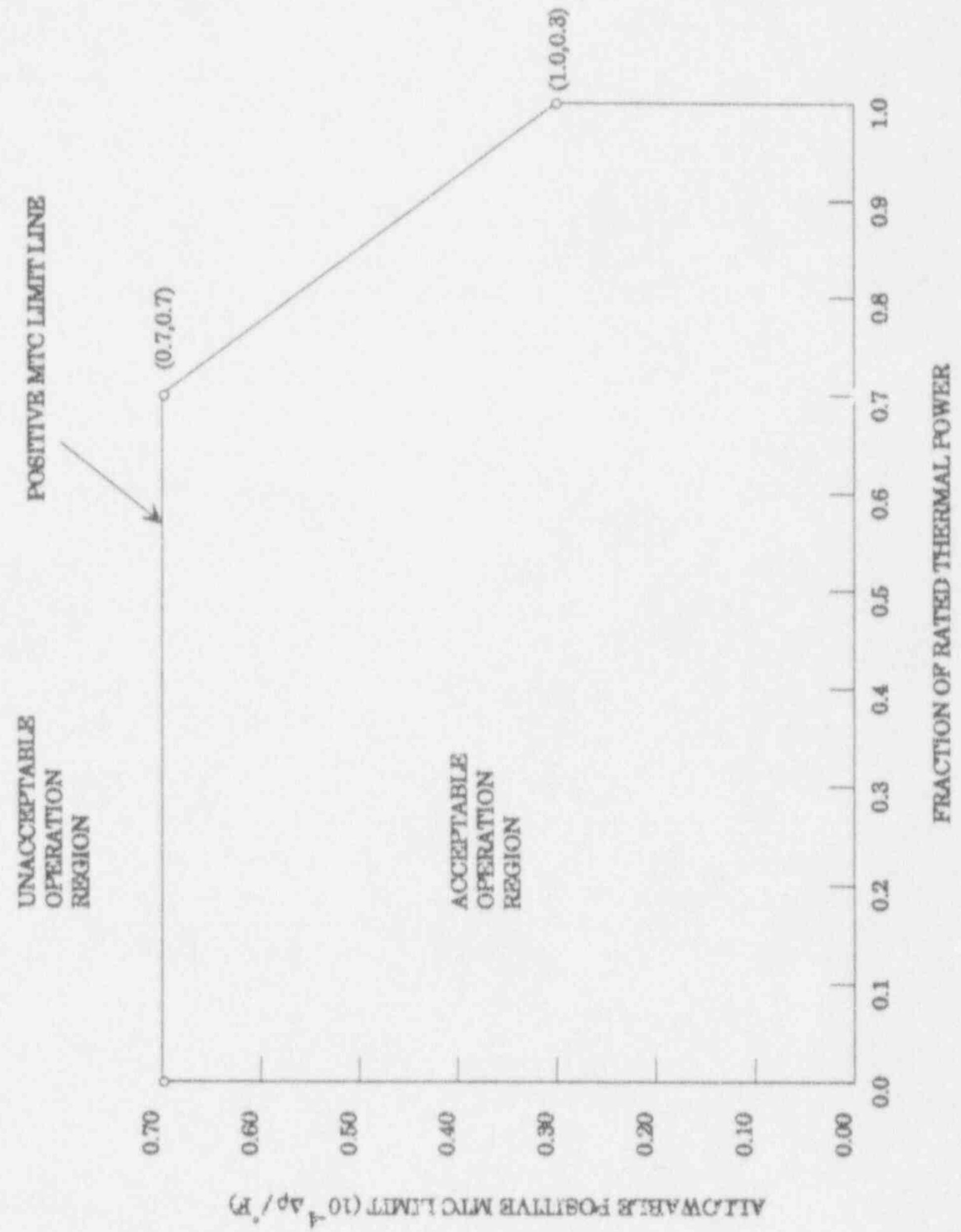

FIGURE 3.1.1-1

FRACTION OF RATED THERMAL POWER
VS. ALLOWABLE POSITIVE MTC LIMIT ($10^{-4}\Delta\rho/F°$)

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 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** above 90% of **RATED THERMAL POWER**, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of **RATED THERMAL POWER**.

c. At any **THERMAL POWER**, within 7 EFPD of reaching a **RATED THERMAL POWER** equilibrium boron concentration of 300 ppm.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.1 BORATION CONTROL

Minimum Temperature For Criticality

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 515°F when the reactor is critical.

APPLICABILITY: **MODES** 1 and 2 with $K_{eff} \geq 1.0$.

ACTION: With a Reactor Coolant System operating loop temperature (T_{avg}) < 515°F, restore T_{avg} to within its limit within 15 minutes or be in **HOT STANDBY** within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 515°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_{avg} is less than 525°F.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Flow Paths - Shutdown

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. A flow path from the boric acid storage tank via either a boric acid pump or a gravity feed connection and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is **OPERABLE**, or

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

APPLICABILITY: **MODES** 5 and 6.

ACTION: With none of the above flow paths **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one injection path is restored to **OPERABLE** status.

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 3.1.2-1 when a flow path from the concentrated boric acid 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.

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* At 365°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 365°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

3/4.1.2 BORATION SYSTEMS

Flow Paths - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. Two flow paths from the boric acid storage tanks required to be **OPERABLE** pursuant to Specifications 3.1.2.8 and 3.1.2.9 via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank 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 that required by Specification 3.1.1.1 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 concentrated boric acid tanks is above the temperature limit line shown on Figure 3.1.2-1.

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.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per **REFUELING INTERVAL** by verifying on a SIAS test signal that:

(1) each automatic valve in the flow path actuates to its correct position, and

(2) each boric acid pump starts.

3/4.1.2 BORATION SYSTEMS

Charging Pump - 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 bus.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With no charging pump or high pressure safety injection pump **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one of the required pumps is restored to **OPERABLE** status.

SURVEILLANCE REQUIREMENTS

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

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* At 365°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 365°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Charging Pumps - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two 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 that required by Specification 3.1.1.1 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 At least two charging pumps shall be demonstrated **OPERABLE**:

a. At least once per **REFUELING INTERVAL** by verifying that each charging pump starts automatically upon receipt of a Safety Injection Actuation Test Signal.

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

* Above 80% **RATED THERMAL POWER** the two **OPERABLE** charging pumps shall have independent power supplies.

3/4.1.2 BORATION SYSTEMS

Boric Acid Pumps - Shutdown

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid 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 above, is **OPERABLE**.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With no boric acid 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 until at least one boric acid pump is restored to **OPERABLE** status.

SURVEILLANCE REQUIREMENTS

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

3/4.1.2 BORATION SYSTEMS

Boric Acid Pumps - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.6

a. The boric acid pump(s) in the boron injection flow path(s) required to be **OPERABLE** pursuant to Specification 3.1.2.2.a shall be **OPERABLE** and capable of being powered from an **OPERABLE** emergency bus.

AND,

When in **MODE** 1 > 80% of **RATED THERMAL POWER**

b. The boric acid pump(s) in the boron injection flow path(s) required to be **OPERABLE** pursuant to Specification 3.1.2.8.a shall be **OPERABLE** and capable of being powered from an **OPERABLE** emergency bus.

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

ACTION: With one boric acid pump required for the boron injection flow path(s) pursuant to either Specification 3.1.2.2.a or 3.1.2.8.a inoperable, restore the boric acid 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 that required by Specification 3.1.1.1 at 200°F; restore the above required boric acid pump(s) to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specifications 4.0.5 and 4.1.2.2.

3/4.1.2 BORATION SYSTEMS

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 storage tank and one associated heat tracing circuit with the tank contents in accordance with Figure 3.1.2-1.

b. The refueling water tank with:

1. A minimum contained borated water volume of 9,844 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With no borated water sources **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one borated water source is restored to **OPERABLE** status.

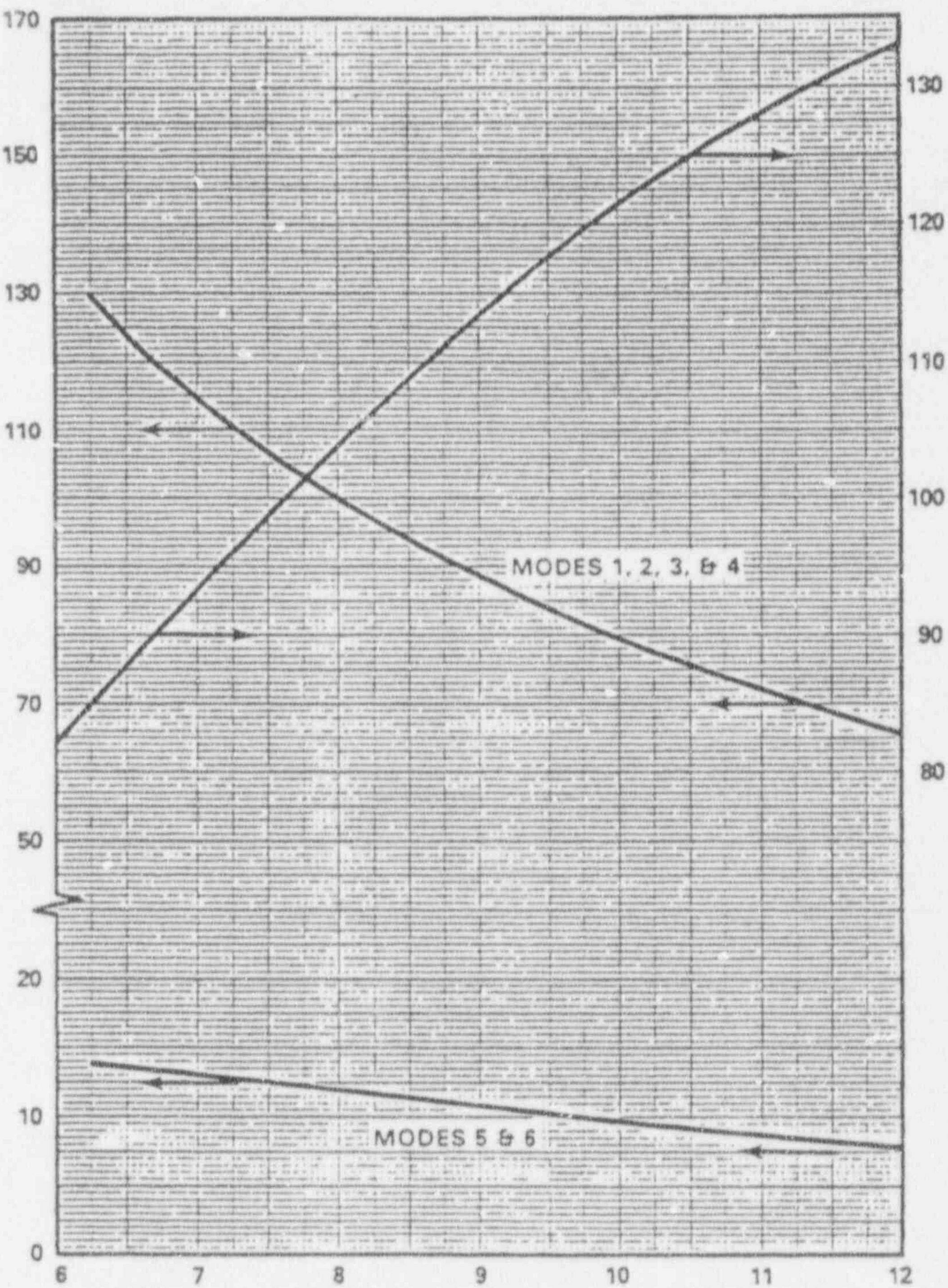
SURVEILLANCE REQUIREMENTS

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 storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is < 35°F.



STORED BORIC ACID CONCENTRATION (WT%)

FIGURE 3.1.2-1

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

3/4.1.2 BORATION SYSTEMS

Charging Pump ECCS Subsystem

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, the following equipment shall be **OPERABLE**:

a. Boric Acid Storage Tank 12 and its associated heat tracing circuit shall be **OPERABLE** per Specification 3.1.2.9.a and the boron injection flow path via Boric Acid Pump 12 from Boric Acid Storage Tank 12 shall be **OPERABLE** per Specification 3.1.2.2.a and Specification 3.1.2.6.

AND,

One of the following:

b. The boron injection flow path from Boric Acid Storage Tank 12 via a gravity feed connection shall be **OPERABLE** per Specification 3.1.2.2.a, or,

Boric Acid Storage Tank 11 and its associated heat tracing circuit shall be **OPERABLE** per Specification 3.1.2.9.a and the boron injection flow path from Boric Acid Storage Tank 11 via a gravity feed connection shall be **OPERABLE** per Specification 3.1.2.2.a.

APPLICABILITY: **MODE** 1 > 80% of **RATED THERMAL POWER**.

ACTION: With only one of the required combinations of borated water sources and flow paths **OPERABLE**, restore two required combinations of borated water sources and flow paths to **OPERABLE** status within 72 hours or reduce power to less than 80% of **RATED THERMAL POWER** within the next 6 hours and comply with Specifications 3.1.2.2, 3.1.2.6, and 3.1.2.9 as applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.8 No additional Surveillance Requirements other than those required by Specifications 4.0.5, 4.1.2.2, and 4.1.2.9

3/4.1.2 BORATION SYSTEMS

Borated Water Sources - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.9 At least two of the following three borated water sources shall be **OPERABLE**:

a. Two boric acid storage tank(s) and one associated heat tracing circuit per tank with the contents of the tanks in accordance with Figure 3.1.2-1 and the boron concentration limited to $\leq$ 8%, and

b. The refueling water tank with:

1. A minimum contained borated water volume of 400,000 gallons,

2. A boron concentration of between 2300 and 2700 ppm,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 100°F in **MODE** 1.

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

ACTION: With only one borated water source **OPERABLE**, restore at least two borated water sources 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 that required by Specification 3.1.1.1 at 200°F; restore at least two borated water sources to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.9 At least two borated water sources shall be demonstrated **OPERABLE**:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume in each water source, and

3. Verifying the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is < 40°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA Position

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Motion Inhibit and all shutdown and regulating CEAs shall be **OPERABLE** with each CEA of a given group positioned within 7.5 inches (indicated position) of all other CEAs in its group.

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

ACTION:

a. With one or more CEAs (regulating or shutdown) inoperable due to being untrippable, be in at least **HOT STANDBY** within 6 hours.

b. With the CEA Motion Inhibit inoperable, within 6 hours either:

   1. Restore the CEA Motion Inhibit to **OPERABLE** status, or

   2. Fully withdraw all CEAs in groups 3 and 4 and withdraw the CEAs in group 5 to less than 5% insertion, or

   3. Be in at least **HOT STANDBY**.

c. With one regulating 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 for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year.

d. With one CEA (regulating or shutdown) inoperable due to causes other than addressed by **ACTION** a, above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in CEA group 5, operation in **MODES** 1 and 2 may continue.

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* See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

e. With one CEA (regulating or shutdown) misaligned from any other CEA in its group by less than 15 inches, operation in **MODES** 1 and 2 may continue, provided that the misaligned CEA is restored to within its specified alignment requirements within one hour, otherwise implement Action g.

f. With one CEA (regulating or shutdown) misaligned from any other CEA in its group by 15 inches or more, operation in **MODES** 1 and 2 may continue, provided that the misaligned CEA is restored to within its specified alignment requirements within the time allowance determined by the full core power distribution monitoring system or, if the full core power distribution monitoring system time allowance is unavailable, the allowable time shall be that provided in the COLR, otherwise implement Action g. If the COLR allowable time is used, the pre-misaligned F_r^T value used to determine the allowable time to realign the CEA shall be the latest measurement taken within 5 days prior to the CEA misalignment. If no measurements were taken within 5 days prior to the misalignment, immediately implement Action g.

LIMITING CONDITION FOR OPERATION (Continued)

g. With one CEA (regulating or shutdown) not within its specified alignment requirements at the conclusion of the time allowance permitted by Actions e, f or h, immediately start to implement the following actions:

   1. If the **THERMAL POWER** level prior to the misalignment was greater than 50% of **RATED THERMAL POWER, THERMAL POWER** shall be reduced to less than the greater of:

      a) 50% of **RATED THERMAL POWER**

      b) 75% of the **THERMAL POWER** level prior to the misalignment

   within one hour after exceeding the time allowance permitted by Actions e, f or h.

   2. If the **THERMAL POWER** level prior to the misalignment was $\leq$ 50% of **RATED THERMAL POWER**, maintain **THERMAL POWER** no higher than the value prior to the misalignment.

   If negative reactivity insertion is required to reduce **THERMAL POWER**, boration shall be used. Within one hour after establishing the appropriate **THERMAL POWER** as required above, either:

   1. Restore the CEA to within its specified alignment requirements, or

   2. Declare the CEA inoperable. After declaring the CEA inoperable, **POWER OPERATION** may continue for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year provided that within one hour after declaring the CEA inoperable, the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 inches of the inoperable CEA while:

      a) maintaining the allowable CEA sequence and CEA group insertion limits provided in the COLR; for a regulating CEA, and with the **THERMAL POWER** level restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

      b) maintaining the remainder of the CEAs in a shutdown group withdrawn to at least 129 inches.

LIMITING CONDITION FOR OPERATION (Continued)

h. With more than one CEA (regulating or shutdown) misaligned and each misaligned CEA is within 15 inches of any other CEA in its group (indicated position) restore the misaligned CEAs to within their specified alignment requirements within one hour, otherwise immediately declare the misaligned CEAs inoperable and implement Action i. If only one CEA (regulating or shutdown) remains misaligned at the end of one hour, implement Action g.

i. With more than one CEA (regulating or shutdown) inoperable or with more than one CEA (regulating or shutdown) misaligned and any one or more of the misaligned CEAs is 15 inches (indicated position) or more from any other CEA in its group, be in at least **HOT STANDBY** within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each CEA not fully inserted shall be determined to be **OPERABLE** by inserting it at least 7.5 inches at least once per 31 days. For the purposes of performing this CEA operability test, if the CEA has an inoperable position indication channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. If a direct position indication (full out reed switch or voltage dividing network) cannot be restored within ten minutes from the commencement of CEA motion, or CEA withdrawal exceeds the surveillance testing insertion by > 7.5 inches, the position of the CEA shall be assumed to have been > 15 inches from its group at the commencement of CEA motion.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated **OPERABLE** at least once per 31 days by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit also prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

Position Indicator Channels

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least two of the following three CEA position indicator channels shall be **OPERABLE** for each shutdown and regulating CEA:

a. CEA voltage divider reed switch position indicator channel, capable of determining the absolute CEA position within $\pm$ 1.75 inches;

b. CEA "Full Out" or "Full In" reed switch position indicator channel, only if the CEA is fully withdrawn or fully inserted, as verified by actuation of the applicable position indicator; and

c. CEA pulse counting position indicator channel.

APPLICABILITY: **MODES** 1 and 2.

ACTION:

a. With a maximum of one CEA per group having its voltage divider reed switch position indicator channel or its pulse counting position indicator channel inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, either:

   1. Within 6 hours

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

      b) Be in at least **HOT STANDBY**, or

      c) Reduce **THERMAL POWER** to $\leq$ 70% of **RATED THERMAL POWER**; if negative reactivity insertion is required to reduce **THERMAL POWER**, boration shall be used. Operation at or below this reduced **THERMAL POWER** level may continue provided that within the next 4 hours either:

         1) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining

LIMITING CONDITION FOR OPERATION (Continued)

the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the **THERMAL POWER** level may be returned to a level consistent with all other applicable specifications and operation may continue per Specification 3.1.3.3 above; or

2) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and **THERMAL POWER** level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6, and may continue per Specification 3.1.3.3 above.

or,

2. If the failure existed before entry into **MODE** 2 or occurs prior to an "all CEAs out" configuration, the CEA group(s) with inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator. These actions must be completed within 10 hours of entry into **MODE** 2 and prior to exceeding 70% of **RATED THERMAL POWER**. The provisions of Specification 3.0.4 are not applicable. Once these actions are completed, operation may continue per Specification 3.1.3.3 above.

b. With more than one CEA per group having its CEA pulse counting position indicator channel and either (1) the "Full Out" or "Full In" position indicator, or (2) the voltage divider position indicator channel inoperable, operation in **MODES** 1 and

LIMITING CONDITION FOR OPERATION (Continued)

2 may continue for up to 24 hours provided that for the affected CEAs, either:

1. The CEA voltage divider reed switch position indicator channels are **OPERABLE**, or
2. The CEA "Full Out" or "Full In" reed switch position indicator channels are **OPERABLE**, with the CEA fully withdrawn or fully inserted as verified by actuation of the applicable position indicator.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each required CEA position indication channel shall be determined to be **OPERABLE** by determining CEA positions as follows at least once per 12 hours, by:

a. Verifying the CEA pulse counting position indicator channels and the CEA voltage divider reed switch position indicator channels agree within 4.5 inches, or

b. Verifying the CEA pulse counting position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches, or

c. Verifying the CEA voltage divider reed switch position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches.

4.1.3.3.2 During time intervals when the deviation circuit is inoperable, the above verification of required CEA position indicator channels shall be made at least once per 4 hours.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA Drop Time

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: **MODES** 1 and 2.

ACTION: 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.

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 of the reactor vessel head,

b. For specifically affected individual 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 **REFUELING INTERVAL**.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

Shutdown CEA Insertion Limit

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: **MODES** 1 and 2 with $K_{eff} \geq 1.0$*.

ACTION: With one or more shutdown CEA(s) withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, consider the CEA(s) misaligned, and immediately apply Specification 3.1.3.1; Action e, f, h, or i, as appropriate.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

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.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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 provided in the COLR (regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches) with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. $\leq$ 4 hours per 24 hour interval,

b. $\leq$ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. $\leq$ 14 Effective Full Power Days per calendar year.

APPLICABILITY: **MODES** 1* and 2 with $K_{eff} \geq 1.0$*.

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, within two 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 limits provided in the COLR.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 4 hours per 24 hour interval, except during operations pursuant to the provisions of **ACTION** items c. and e. of Specification 3.1.3.1, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits provided in the COLR are not exceeded, or

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* See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FCR OPERATION (Continued)

2. Any subsequent increase in **THERMAL POWER** is restricted to $\leq$ 5% of **RATED THERMAL POWER** per hour.

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of **ACTION** items c. and e. of Specification 3.1.3.1, 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 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 Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 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 provided in the COLR.

APPLICABILITY: **MODE 1.**

ACTION: With the linear heat rate exceeding the limits provided in the COLR, as indicated by four or more coincident incore channels or by the **AXIAL SHAPE INDEX** outside of the power dependent control limits provided in the COLR, 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 one 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 by continuously monitoring the core power distribution with either the Excore Detector Monitoring System or with the Incore Detector Monitoring System.

4.2.1.3 Excore Detector Monitoring System - The Excore Detector Monitoring System may be used for monitoring the core power distribution* by:

a. Verifying at least once per 12 hours that the CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. Verifying at least once per 31 days that the **AXIAL SHAPE INDEX** alarm setpoints are adjusted to within the limits provided in the COLR.

* See Surveillance 4.2.2.1.2.d.

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the **AXIAL SHAPE INDEX** is maintained within the limits provided in the COLR, where 100 percent of the allowable power is the maximum allowable fraction of **RATED THERMAL POWER** as determined by the F^T_{xy} limit provided in the COLR.

4.2.1.4 Incore Detector Monitoring System - The Incore Detector Monitoring System may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days* of accumulated operation in **MODE** 1.

b. Have their alarm setpoint adjusted to protect the linear heat rate limit provided in the COLR when the uncertainty factors provided in the COLR** are appropriately included in the setting of these alarms.

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* For Unit 1 Cycle 11 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, this surveillance shall be performed at least once per 15 days of accumulated operation in **MODE** 1.

** For Unit 1 Cycle 11 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, the measurement-calculational uncertainty factor on linear heat rate shall be increased by 1% (from 1.062 to 1.072) prior to comparison with the Technical Specification limits.

3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2.1 The calculated value of F^T_{xy} shall be within the limit provided in the COLR.**

APPLICABILITY: **MODE** 1*.

ACTION: With F^T_{xy} outside the limit provided in the COLR, within 6 hours either:

a. Withdraw and maintain the CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce **THERMAL POWER** as follows:

    1. Reduce **THERMAL POWER** to bring the combination of **THERMAL POWER** and F^T_{xy} within the limits provided in the COLR, or

    2. Reduce **THERMAL POWER** to less than or equal to the limit established by the full core power distribution monitoring system as a function of F^T_{xy}; or

b. Be in at least **HOT STANDBY**.

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

b. At least once per 31 days*** of accumulated operation in **MODE** 1, and

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** For Unit 1 Cycle 11 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, the calculated value of F^T_{xy} shall be increased by 1% prior to comparison with the limit.

* See Special Test Exception 3.10.2.

*** For Unit 1 Cycle 11 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, this surveillance shall be performed at least once per 15 days of accumulated operation in **MODE** 1.

SURVEILLANCE REQUIREMENTS (Continued)

c. Within four hours if the **AZIMUTHAL POWER TILT** (T_q) is outside the limit provided in Specification 3.2.4.

d. At least once per 3 days of accumulated operation in **MODE** 1 when monitoring linear heat rate using the Excore Detector Monitoring System per Surveillance 4.2.1.3.

4.2.2.1.3 F^T_{xy} shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all CEAs at or above the Long Term Steady State Insertion Limit. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

3/4.2.3 TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be within the limits provided in the COLR.**

APPLICABILITY: **MODE 1***.

ACTION: With F_r^T outside the limits provided in the COLR, within 6 hours either:

a. Be in at least **HOT STANDBY**, or

b. Withdraw and maintain the CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce **THERMAL POWER** as follows:

   1. Reduce **THERMAL POWER** to bring the combination of **THERMAL POWER** and F_r^T within the limits provided in the COLR and maintain the peripheral axial shape index within the DNB axial flux offset control limits provided in the COLR, or

   2. Reduce **THERMAL POWER** to less than or equal to the limit established by the full core power distribution monitoring system as a function of F_r^T.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

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** For Unit 1 Cycle 11 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, the calculated value of F_r^T shall be increased by 1% prior to comparison with the limit.

* See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days*** of accumulated operation in **MODE** 1, and

c. Within four hours if the **AZIMUTHAL POWER TILT** (T_q) is outside the limit given in Specification 3.2.4.

4.2.3.3 F_r^T shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all CEAs at or above the Long Term Steady State Insertion Limit.

*** For Unit 1 Cycle 11 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, this surveillance shall be performed at least once per 15 days of accumulated operation in **MODE** 1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.4 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The **AZIMUTHAL POWER TILT** (T_q) shall not exceed 0.030.

APPLICABILITY: **MODE** 1 above 50% of **RATED THERMAL POWER***.

ACTION:

a. With the indicated **AZIMUTHAL POWER TILT** determined to be > 0.030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the **TOTAL PLANAR RADIAL PEAKING FACTOR** (F^T_{xy}) and the **TOTAL INTEGRATED RADIAL PEAKING FACTOR** (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated **AZIMUTHAL POWER TILT** determined to be > 0.10, operation may proceed for up to 2 hours provided that the **TOTAL INTEGRATED RADIAL PEAKING FACTOR** (F^T_r) and **TOTAL PLANAR RADIAL PEAKING FACTOR** (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the **THERMAL POWER** level is restricted to ≤ 20% of **RATED THERMAL POWER**.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The **AZIMUTHAL POWER TILT** shall be determined to be within the limit by:

a. Calculating the tilt at least once per 12 hours, and

b. Using the incore detectors to determine the **AZIMUTHAL POWER TILT** at least once per 12 hours when one excore channel is inoperable and **THERMAL POWER** is > 75% of **RATED THERMAL POWER**.

* See Special Test Exception 3.10.2.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown:

a. Cold Leg Temperature ≤ 548°F

b. Pressurizer Pressure ≥ 2200 psia*

c. Reactor Coolant System Total Flow Rate ≥ 370,000 gpm

d. **AXIAL SHAPE INDEX, THERMAL POWER** as specified in the COLR.

APPLICABILITY: **MODE** 1.

ACTION: With any of the above parameters exceeding its limit, restore the parameter 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

4.2.5.1 Each of the parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

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* Limit not applicable during either a **THERMAL POWE** amp increase in excess of 5% of **RATED THERMAL POWER** per minute or a **ERMAL POWER** step increase of greater than 10% of **RATED THERMAL POW**

Incore Detectors

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Incore Detection System shall be **OPERABLE** with at least one **OPERABLE** detector segment in each core quadrant on each of the four axial elevations containing incore detectors and as further specified below:

a. For monitoring the **AZIMUTHAL POWER TILT**:*

At least two quadrant symmetric incore detector segment groups at each of the four axial elevations containing incore detectors in the outer 184 fuel assemblies with sufficient **OPERABLE** detector segments in these detector groups to compute at least two **AZIMUTHAL POWER TILT** values at each of the four axial elevations containing incore detectors.

b. For recalibration of the Excore Neutron Flux Detector System:

1. At least 75%** of all incore detector segments,

2. A minimum of 9 **OPERABLE** incore detector segments at each detector segment level, and

3. A minimum of 2 **OPERABLE** detector segments in the inner 109 fuel assemblies and 2 **OPERABLE** segments in the outer 108 fuel assemblies at each segment level.

c. For monitoring the **TOTAL PLANAR RADIAL PEAKING FACTOR**, the **TOTAL INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate:

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* For Unit 1 Cycle 11 only, the following requirements shall be substituted for Limiting Condition for Operation 3.3.3.2.a:

At least eight quadrant symmetric incore detector segment groups containing incore detectors in the outer 184 fuel assemblies with sufficient **OPERABLE** detector segments in these detector groups to compute at least one **AZIMUTHAL POWER TILT** value at each of the four axial elevations containing incore detectors and at least two **AZIMUTHAL POWER TILT** values at three axial elevations containing incore detectors.

** For Unit 1 Cycle 11 only, the following requirement shall be substituted for Limiting Condition for Operation 3.3.3.2.b.1:

At least 60% of all incore detector segments,

LIMITING CONDITION FOR OPERATION (Continued)

1. At least 75%*** of all incore detector locations,

2. A minimum of 9 **OPERABLE** incore detector segments at each detector segment level, and

3. A minimum of 2 **OPERABLE** detector segments in the inner 109 fuel assemblies and 2 **OPERABLE** segments in the outer 108 fuel assemblies at each segment level. An **OPERABLE** incore detector segment shall consist of an **OPERABLE** rhodium detector constituting one of the segments in a fixed detector string.An **OPERABLE** incore detector location shall consist of a string in which at least three of the four incore detector segments are **OPERABLE**.

An **OPERABLE** quadrant symmetric incore detector segment group shall consist of a minimum of three **OPERABLE** rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the Incore Detection System is used for:

a. Monitoring the **AZIMUTHAL POWER TILT**,

b. Recalibration of the Excore Neutron Flux Detection System, or

c. Monitoring the **TOTAL PLANAR RADIAL PEAKING FACTOR**, the **TOTAL INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

ACTION: With the Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

*** For Unit 1 Cycle 11 only, the following requirement shall be substituted for Limiting Condition for Operation 3.3.3.2.c.1:

At least 60% of all incore detector locations,

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:

1. Monitoring the **AZIMUTHAL POWER TILT**.

2. Recalibration of the Excore Neutron Flux Detection System.

3. Monitoring the **TOTAL PLANAR RADIAL PEAKING FACTOR**, the **TOTAL INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

b. At least once per **REFUELING INTERVAL** 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.

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.***

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 Power Level-High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With one reactor coolant loop and associated steam generator in operation and with one or more main steam line code safety valves associated with the operating steam generator inoperable, operation in **MODE** 3 may proceed provided:

   1. That at least 2 main steam line code safety valves on the non-operating steam generator are **OPERABLE**, and

   2. That within 4 hours the inoperable valve is restored to **OPERABLE** status; otherwise, be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

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* Entry into **MODE** 3 is permitted to determine **OPERABILITY** of main steam line code safety valves. During this time, at least 2 main steam line code safety valves per steam generator shall be **OPERABLE**.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT (PERCENT OF RATED THERMAL POWER)
1	93
2	79
3	66

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE	LIFT SETTINGS* ALLOWABLE	ORIFICE SIZE
a. RV-3992/4000	935-995 psig	R
b. RV-3993/4001	935-995 psig	R
c. RV-3994/4002	935-1035 psig	R
d. RV-3995/4003	935-1035 psig	R
e. RV-3996/4004	935-1065 psig	R
f. RV-3997/4005	935-1065 psig	R
g. RV-3998/4006	935-1065 psig	R
h. RV-3999/4007	935-1065 psig	R

* Lift settings for a given steam line are also acceptable if any 2 valves lift between 935 and 995 psig, any 2 other valves lift between 935 and 1035 psig, and the 4 remaining valves lift between 935 and 1065 psig.

3/4.7.1 TURBINE CYCLE

Auxiliary Feedwater System

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two auxiliary feedwater trains consisting of one steam-driven and one motor-driven pump and associated flow paths capable of automatically initiating flow shall be **OPERABLE**. (An **OPERABLE** steam-driven train shall consist of one pump aligned for automatic flow initiation and one pump aligned in standby.)*

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

ACTION:

a. With any single pump inoperable, perform the following:

1. With No. 13 motor-driven pump inoperable:

(a) Align the standby steam-driven pump to automatic initiating status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours, and

(b) Restore No. 13 motor-driven pump to **OPERABLE** status within the next 7 days or be in **HOT SHUTDOWN** within the next 12 hours.

2. With one steam-driven pump inoperable:

(a) Align the **OPERABLE** steam-driven pump to automatic initiating status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours, and

(b) Restore the inoperable steam-driven pump to standby status (or automatic initiating status if the other steam-driven pump is to be placed in standby) within the next 7 days or be in **HOT SHUTDOWN** within the next 12 hours.

b. With any two pumps inoperable:

1. Verify that the remaining pump is aligned to automatic initiating status within one hour, and

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* A standby pump shall be available for operation but aligned so that automatic flow initiation is defeated upon AFAS actuation.

LIMITING CONDITION FOR OPERATION (Continued)

2. Verify within one hour that No. 23 motor-driven pump is **OPERABLE** and valve 2-CV-4550 has been exercised within the last 30 days, and

3. Restore a second pump to automatic initiating status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours.

c. Whenever a subsystem(s) (a subsystem consisting of one pump, piping, valves and controls in the direct flow path) required for **OPERABILITY** is inoperable for the performance of periodic testing (e.g., manual discharge valve closed for pump Total Dynamic Head Test or Logic Testing) a dedicated operator(s) will be stationed at the local station(s) with direct communication to the Control Room. Upon completion of any testing, the subsystem(s) required for **OPERABILITY** will be returned to its proper status and verified in its proper status by an independent operator check.

d. The requirements of Specification 3.0.4 are not applicable whenever one motor and one steam-driven pump (or two steam-driven pumps) are aligned for automatic flow initiation.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater flowpath shall be demonstrated **OPERABLE**:

a. At least once per 31 days by:

1. Verifying that each steam-driven pump develops a Total Dynamic Head of $\geq$ 2800 ft. on recirculation flow (if verification must be demonstrated during **STARTUP**, surveillance testing shall be performed upon achieving an RCS temperature $\geq$ 300°F and prior to entering **MODE** 1).

2. Verifying that the motor-driven pump develops a Total Dynamic Head of $\geq$ 3100 ft. on recirculation flow.

3. Cycling each testable, remote-operated valve that is not in its operating position through at least one complete cycle.

4. Verifying that each valve (manual, power-operated or automatic) in the direct flow path that is not locked, sealed or otherwise secured in position is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

b. Before entering **MODE** 3 after a **COLD SHUTDOWN** of at least 14 days by completing a flow test that verifies the flow path from the condensate storage tank to the steam generators.

c. At least once per 18 months by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position (verification of flow-modulating characteristics not required) and each auxiliary feedwater pump automatically starts upon receipt of each AFAS test signal, and

   2. Verifying that the Auxiliary Feedwater System is capable of providing a minimum of 300 gpm nominal flow to each flow leg.*

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* This surveillance may be performed on one flow leg at a time.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

Condensate Storage Tank

LIMITING CONDITION FOR OPERATION

3.7.1.3 The No. 12 condensate storage tank (CST) shall be **OPERABLE** with a minimum contained water volume of 150,000 gallons per unit.

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

ACTION: With the No. 12 condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to **OPERABLE** status or be in **HOT SHUTDOWN** within the next 12 hours, or

b. Demonstrate the **OPERABILITY** of the No. 11 condensate storage tank as a backup supply to the auxiliary feedwater pumps and restore the No. 12 condensate storage tank to **OPERABLE** status within 7 days or be in **HOT SHUTDOWN** within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The No. 12 condensate storage tank shall be demonstrated **OPERABLE** at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The No. 11 condensate storage tank shall be demonstrated **OPERABLE** at least once per 12 hours by verifying that the tank contains a minimum of 150,000 gallons of water and by verifying that the flow path for taking suction from this tank is **OPERABLE** with the manual valves in this flow path open whenever the No. 11 condensate storage tank is the supply source for the auxiliary feedwater pumps.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

Activity

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be ≤ 0.10 μCi/gram **DOSE EQUIVALENT I-131**.

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

ACTION: With the specific activity of the Secondary Coolant System > 0.10 μCi/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 4.7-2.

TABLE 4.7-2

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.

3/4.7 **PLANT SYSTEMS**

3/4.7.1 TURBINE CYCLE

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 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, **POWER OPERATION** may continue provided the inoperable valve is either restored to **OPERABLE** status or closed within 4 hours; otherwise, be in **HOT SHUTDOWN** within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in **MODES** 1, 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in **HOT SHUTDOWN** within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated **OPERABLE** by verifying full closure in less than 5.2 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 80°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION: With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant < 80°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two component cooling water loops shall be **OPERABLE**. At least one component cooling water heat exchanger shall be operating and the remaining component cooling water heat exchanger may be in standby.

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

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

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops 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 **REFUELING INTERVAL** during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation test signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent service water loops shall be **OPERABLE.**

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

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

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two service water loops 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 **REFUELING INTERVAL** during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on Safety Injection Actuation and Containment Spray Actuation test signals.

3/4.7.5 SALTWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 At least two independent saltwater loops shall be **OPERABLE**.

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

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 At least two saltwater loops shall be **OPERABLE**.

a. At least once per 31 days be 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 **REFUELING INTERVAL** during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation test signal.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The Control Room Emergency Ventilation System shall be **OPERABLE** with:

a. Two filter trains,

b. Two air conditioning units,

c. Two isolation valves in each Control Room outside air intake duct,

d. Two isolation valves in the common exhaust to atmosphere duct, and

e. One isolation valve in the toilet area exhaust duct.

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

ACTION:

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

b. With one air conditioning unit inoperable, restore the inoperable unit 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.

c. With one isolation valve per Control Room outside air intake duct inoperable, operation may continue provided the other isolation valve in the same duct is maintained closed; otherwise, be in at least **HOT STANDBY** within 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

d. With one common exhaust to atmosphere duct isolation valve inoperable, restore the inoperable valve 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.

e. With the toilet area exhaust duct isolation valve inoperable, restore the inoperable valve 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

4.7.6.1 The Control Room Emergency Ventilation System shall be demonstrated **OPERABLE**:

a. At least once per 62 days, on a **STAGGERED TEST BASIS**, by deenergizing the backup Control Room air conditioner and verifying that the emergency Control Room air conditioners maintain the air temperature $\leq$ 104°F for at least 12 hours when in the recirculation mode.

b. At least once per 31 days by initiation flow through each HEPA filter and charcoal adsorber train and verifying that each train operates for at least 15 minutes.

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

   1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

   2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

   3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying a system flow rate of 2000 cfm $\pm$ 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 from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the filter train shall be demonstrated **OPERABLE** by also verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow of 2000 cfm $\pm$ 10%.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

2. Verifying that on a Control Room high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks and that both of the isolation valves in each inlet duct and common exhaust duct, and the isolation valve in the toilet area exhaust duct, close.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter system at a flow rate of 2000 cfm $\pm$ 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter system at a flow rate of 2000 cfm $\pm$ 10%.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The ECCS Pump Room Exhaust Ventilation System shall be **OPERABLE** with one HEPA filter and charcoal adsorber train and two exhaust fans.

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

ACTION:

a. With one ECCS pump room exhaust fan inoperable, restore the inoperable fan 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.

b. With the ECCS exhaust filter train inoperable, restore the filter train 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

4.7.7.1 The ECCS Pump Room Exhaust Ventilation System shall be demonstrated **OPERABLE**:

a. At least once per 31 days by initiating, from the Control Room, flow through the HEPA filter and charcoal adsorber train and verifying that each exhaust fan operates for at least 15 minutes.

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 charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

4. 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 from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the filter train shall be demonstrated **OPERABLE** by also verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow rate of 3000 cfm $\pm$ 10%.

d. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

g. After maintenance affecting the air flow distribution by testing in-place and verifying that the air flow distribution is uniform within $\pm$ 20% of the average flow per unit when tested in accordance with the provisions of Section 9 of "Industrial Ventilation" and Section 8 of ANSI N510-1975.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All safety related snubbers[1] shall be **OPERABLE**.

APPLICABILITY: **MODES** 1, 2, 3 and 4. (**MODES** 5 and 6 for snubbers located on systems required **OPERABLE** in those **MODES**.)

ACTION: With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to **OPERABLE** status, and perform an engineering evaluation* per Specification 4.7.8.1.b and c on the supported component or declare the supported system inoperable and follow the appropriate **ACTION** statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each snubber shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5. As used in this Specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

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1 Safety related snubbers include those snubbers installed on safety related systems and snubbers on non-safety related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety related system.

* A documented, visual inspection shall be sufficient to meet the requirements for an engineering evaluation. Additional analyses, as needed, shall be completed in a reasonable period of time.

SURVEILLANCE REQUIREMENTS (Continued)

a. Visual inspections

Visual inspections shall be performed in accordance with the schedule determined by Table 4.7-3. Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently or jointly according to the schedule determined by Table 4.7-3. The visual inspection interval for each population or category of snubbers shall be determined based upon the criteria provided in Table 4.7-3 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 159.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired **OPERABILITY**, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. 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, remedied and functionally tested for that particular snubber and for other snubbers that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined **OPERABLE** per Specification 4.7.8.1.d, as applicable. When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined **OPERABLE** via functional testing for the purpose of establishing the next visual inspection interval.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

c. Functional Tests

At least once per **REFUELING INTERVAL**, a representative sample of 10% of each type of snubbers in use in the plant shall be functionally tested either in-place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8.1.d, an additional 5% of that type snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

Snubbers identified as "Especially Difficult to Remove" or in "High Exposure Zones" shall also be included in the representative sample.**

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period. Failure of these snubbers shall not entail functional testing of additional snubbers.

** Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber **OPERABILITY** for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

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 generically susceptible snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

SURVEILLANCE REQUIREMENTS (Continued)

At least once per **REFUELING INTERVAL****, the installation and maintenance records for each safety related snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review.* If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

** The first snubber service life review following Amendment No. 165 shall be performed within 18 months* of the previous review.

* The provisions of Specification 4.0.2 are applicable.

TABLE 4.7-3

SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF INOPERABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

TABLE 4.7-3 (Continued)

TABLE NOTATION

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of inoperable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of inoperable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of inoperable snubbers as determined by interpolation.

Note 3: If the number of inoperable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of inoperable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of inoperable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of inoperable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of inoperable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 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 ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.9.1.2 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.

SURVEILLANCE REQUIREMENTS (Continued)

a. Sources in use - At least once per six 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.

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

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

4.7.9.1.3 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 $\geq$ 0.005 microcuries of removable contamination.

3/4.7.10 WATERTIGHT DOORS

LIMITING CONDITION FOR OPERATION

3.7.10 The following watertight doors shall be closed except when the door is being used for normal entry and exit:

a. ECCS Pump Room Doors (4).

b. Service Water Pump Room to Heater Bay Doors (2).

c. Auxiliary Feed Pump Room to Heater Bay Doors (2).

d. Emergency Escape Hatch, Service Water Pump Room from Penetration Room.

e. Main Steam Piping Area from Piping Penetration Room Door.

f. Passage to Main Steam Piping Area Door.

g. Warehouse to Intake Structure Door, Elevation 12'.

h. Outside to Intake Structure Door.

i. Warehouse to Intake Structure Door Elevation 29'.

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

ACTION: With one or more of the above doors open, restore the door to its closed position 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

4.7.10 The above watertight doors shall be determined closed at least once per 12 hours.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Fire Suppression Water System

LIMITING CONDITION FOR OPERATION

3.7.11.1 The Fire Suppression Water System shall be **OPERABLE** with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Two water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An **OPERABLE** flow path capable of taking suction from the Pretreated Water Storage Tanks Numbers 11 and 12 and transferring the water through distribution piping with **OPERABLE** sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser required to be **OPERABLE** per Specifications 3.7.11.2, 3.7.11.4, and 3.7.11.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to **OPERABLE** status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable:

1. Establish a backup Fire Suppression Water System within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.9.2:

   a) By telephone within 24 hours,

LIMITING CONDITION FOR OPERATION (Continued)

b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

c) In writing 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.

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The Fire Suppression Water System shall be demonstrated **OPERABLE**:

a. At least once per 7 days be verifying the contained water supply volume.

b. At least once per 31 days on a **STAGGERED TEST BASIS** by starting the electric motor driven pump and operating it for at least 15 minutes. This test shall be performed on a **STAGGERED TEST BASIS** with the test required by 4.7.11.1.2.a.2.

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.

d. At least once per 12 months by performance of a system flush of the filled portions of the system.

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 automatic valve in the flow path actuates to its correct position,

2. Verifying that each pump develops at least 2500 gpm at a discharge pressure of 125 psig,

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each high pressure pump starts (sequentially) to maintain the Fire Suppression Water System pressure ≥ 80 psig.

g. At least once per **REFUELING INTERVAL** by: (1) 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, and (2) performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

4.7.11.1.2 The 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 174 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes. This test shall be performed on a **STAGGERED TEST BASIS** with the test required by Specification 4.7.11.1.1.b.

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-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for viscosity, water and sediment.

c. At least once per 18 months by:

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

2. Verifying the diesel starts from ambient conditions on the auto-start signal and operates for ≥ 20 minutes while loaded with the fire pump.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.1.3 The fire pump diesel starting 24-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 $\geq$ 24 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 anti-corrosion material.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Spray and/or Sprinkler Systems

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems shown in Table 3.7-5 shall be **OPERABLE**:

APPLICABILITY: Whenever equipment in the spray/sprinkler protected areas is required to be **OPERABLE**.

ACTION:

a. With one or more of the required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant safe shutdown systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 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, not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 12 months by cycling each 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 verifying that the automatic valves in the flow path actuate to their correct positions on a simulated test signal.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the area in the vicinity of each nozzle(s) to verify the spray pattern will not be obstructed.

TABLE 3.7-5

FIRE PROTECTION SPRINKLERS
UNIT 1

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
11 Diesel Generator	45'-0"
12 Diesel Generator	45'-0"
Unit 1 East Pipe Pen Room 227/316*	5'-0"
Unit 1 Aux Feed Pump Room 603*	12'-0"
Unit 1 East Piping Area Room 428*	45'-0"
Unit 1 East Electrical Penetration Room 429*	45'-0"
Unit 1 West Electrical Penetration Room 423*	45'-0"
Unit 1 Main Steam Piping Room 315*	45'-0"
Unit 1 Component Cooling Pump Room 228*	5'-0"
Unit 1 East Piping Area 224*	5'-0"
Unit 1 Radiation Exhaust Vent Equipment Room 225*	5'-0"
Unit 1 Service Water Pump Room 226*	5'-0"
Unit 1 Boric Acid Tank and Pump Room 217*	5'-0"
Unit 1 Reactor Coolant Makeup Pump Room 216*	5'-0"
Unit 1 Charging Pump Room 115*	(-)10'-0"
Unit 1 Misc Waste Mon Room 113*	(-)10'-0"
Cask and Eqpt Loading Area Rooms 419, 420, 425 & 426*	45'-0"
Solid Waste Processing*	45'-0"
Corridors 200, 202, 212 and 219*	5'-0
Corridors 100, 103 and 116*	(-)10'-0"
Cable Chase 1A*	45'-0"
Cable Chase 1B*	45'-0"
Unit 1 ECCS Pump Room 119*	(-)15'-0"
Hot Instrument Shop Room 222*	5'-0"
Hot Machine Shop Room 223*	5'-0"

* Sprinklers required to ensure the **OPERABILITY** of redundant safe shutdown equipment.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Halon Systems

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following Halon Systems shall be **OPERABLE** with the storage tanks having at least 95% of full charge weight (or level) and 90% of full charge pressure.

a. Cable spreading room total flood system, and associated vertical cable chase 1C, Unit 1.

b. 4160 volt switchgear room 27' & 45' elevation Unit 1.

APPLICABILITY: Whenever equipment protected by the Halon System is required to be **OPERABLE**.

ACTION:

a. With both the primary and backup Halon Systems protecting the areas inoperable, within one hour establish an hourly fire watch with backup fire suppression equipment for those areas protected by the inoperable Halon System. Restore the system to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3 Each of the above required Halon 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 is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 12 months by performing a visual inspection of the nozzle(s) and visible flow paths for obstructions.

d. At least once per 18 months by verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

e. Following completion of major maintenance or modifications on the system(s), within 72 hours by performance of a flow test through headers and nozzles to assure no blockage.

3/4.7 **PLANT SYSTEMS**

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Fire Hose Stations

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-6 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 3.7-6 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an **OPERABLE** hose station within 1 hour. Restore the fire hose station(s) to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the fire hose station(s) to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated **OPERABLE**:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station. Hose stations located in the containment shall be visually inspected on each scheduled reactor shutdown, but not more frequently than every 31 days.

b. At least once per 18 months for hose stations located outside the containment and once per **REFUELING INTERVAL** for hose stations inside the containment by:

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

   2. Replacement of all degraded gaskets in couplings.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years for hose stations located outside the containment and once per **REFUELING INTERVAL** for hose stations inside the containment 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 at least 50 psig greater than the maximum pressure available at that hose station or replacement with a new hose.

TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION	NUMBER OF HOSE STATIONS
1. Containment	10'	2
	45'	2
	69'	2
2. Auxiliary Building	-15'*	1**
	-10'*	2**
	5'	6
	27'	3
	45'	5
	69'*	4
3. Turbine Building, Heater Bay Outside Service Water Pump Rooms and Aux Feedwater Pump Rooms	12'	3
Outside Switchgear Room	27'	2
Outside Switchgear Room	45'	3
4. Intake Structure	10'*	1

* Fire Hose Stations required for primary protection to ensure the **OPERABILITY** of safety related equipment.

** Hose Stations which serve both Units 1 and 2.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Yard Fire Hydrants and Hydrant Hose Houses

LIMITING CONDITION FOR OPERATION

3.7.11.5 The following yard fire hydrants and associated hydrant hose houses shall be **OPERABLE**:

a. #6 yard hydrant and associated hydrant hose house, which provides primary protection for Unit 2 RWT blockhouse.

b. #7 yard hydrant and associated hydrant hose house, which provides primary protection for Unit 1 RWT blockhouse.

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 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 hydrant or associated hydrant hose house is the primary means of fire suppression. Restore the hydrant or hose house to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the hydrant or hose house to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.5 Each of the yard fire hydrants and associated hydrant hose houses 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 6 months (once during March, April or May and once during September, October or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any yard fire hydrant.

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**.

3/4.7.12 PENETRATION FIRE BARRIERS

LIMITING CONDITION FOR OPERATION

3.7.12 All fire barrier penetrations (i.e., cable penetration barriers, firedoors and fire dampers), in fire zone boundaries, protecting safe shutdown areas shall be **OPERABLE**.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations inoperable within one hour either establish a continuous fire watch on at least one side of the affected penetration, or verify the **OPERABILITY** of fire detectors on at least one side of the inoperable fire barrier and establish an hourly fire watch patrol; or verify the operability of Automatic Sprinkler Systems (including the water flow alarm and supervisory system) on both sides of the inoperable fire barrier. Restore the inoperable fire barrier penetration(s) to **OPERABLE** status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperable penetration and plans and schedule for restoring the fire barrier penetration(s) to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.12 Each of the above required fire barrier penetrations shall be verified to be **OPERABLE**:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a fire barrier penetration to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration(s).

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling pool shall be maintained within the limit provided in the COLR.

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 $\geq$ 40 gpm of 2300 ppm boric acid solution or its equivalent until the boron concentration is within its limit. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration shall be determined to be within its limit 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.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling pool shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

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.

The most limiting **SHUTDOWN MARGIN** requirement at beginning of cycle is determined by the requirements of several transients, including Boron Dilution and Steam Line Rupture. The **SHUTDOWN MARGIN** requirements for these transients are relatively small and nearly the same. However, the most limiting **SHUTDOWN MARGIN** requirement at end of cycle comes from just one transient, the Steam Line Rupture event. The requirement for this transient at end of cycle is significantly larger than that for any other event at that time in cycle and, also, considerably larger than the most limiting requirement at beginning of cycle.

Adherence to Technical Specification 3.1.1.1 provides assurance that the available **SHUTDOWN MARGIN** at anytime in cycle will exceed the most limiting **SHUTDOWN MARGIN** requirement at that time in cycle. Without the specified **SHUTDOWN MARGIN** available, immediate boration is required (by Specifications 3/4.1.1.1 or 3/4.1.1.2) that is at least equivalent to boration from the refueling water tank, at its minimum boric acid concentration, via a charging pump, at its minimum flow rate. For example, lower flow rates with higher boric acid concentration could also provide the equivalent boration, but should be verified as equivalent prior to use. When in **MODE** 1 or 2 with $K_{eff} \geq 1.0$, operation within the CEA Group Insertion Limits assures that there is sufficient available **SHUTDOWN MARGIN** to match the **SHUTDOWN MARGIN** requirements in the safety analysis.

In **MODE** 5, the reactivity transients resulting from any event are minimal and do not vary significantly during the cycle. The most limiting event at any time during the cycle is a Boron Dilution Event with the pressurizer level less than 90 inches and the sources of non-borated water restricted. Adherence to Technical Specification 3.1.1.2 provides assurance that the available **SHUTDOWN MARGIN** will exceed the most limiting **SHUTDOWN MARGIN** requirement at any time in cycle.

3/4.1.1.3 Boron Dilution

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,601 cubic feet in approximately 24 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 Moderator Temperature Coefficient (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses 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.

3/4.1.1.5 Minimum Temperature For Criticality

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°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, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boration System is a subset of the Chemical Volume and Control System. The Boration System ensures that negative reactivity control is available during each **MODE** of facility operation. The system also provides coolant flow following a SIAS (e.g., during a Small Break LOCA) to supplement flow from the Safety Injection System. Above 80% of **RATED THERMAL POWER**, the Small Break LOCA analyses assume flow from a single charging pump, accounting for measurement uncertainties and flow mal-distribution effects in calculating a conservative value of charging flow actually delivered to the RCS. Credit is only taken for the water inventory, no credit is taken for the injected boron. Above 80% of **RATED THERMAL POWER**, two independent, redundant, and automatic boration systems are provided to ensure functional capability in the event an assumed failure renders one of the systems inoperable.

The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from **OPERABLE** diesel generators. At or below 80% of **RATED THERMAL POWER**, there is a corresponding decrease in decay heat which compensates for the loss of injection from one charging pump assumed in the Small Break LOCA analyses.

3/4.1 <u>REACTIVITY CONTROL SYSTEMS</u>

BASES

With the RCS average temperature above 200°F, a minimum of two independent and redundant boration 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 will provide sufficient **SHUTDOWN MARGIN** from all operating conditions assuming xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid tanks, the concentration and volume of which are met by the range of values given in Specifications 3.1.2.8 and 3.1.2.9, or 55,627 gallons of 2300 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during **MODES** 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

With the RCS temperature below 200°F, one boration 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 change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing sufficient **SHUTDOWN MARGIN** after xenon decay and cooldown from 200°F to 140°F. This condition requires either boric acid solution from the boric acid tanks, the requirements of which are met by Specification 3.1.2.7, or 9,844 gallons of 2300 ppm borated water from the refueling water tank.

The **OPERABILITY** of one boration system during **REFUELING** ensures that this system is available for reactivity control while in **MODE** 6.

3/4.1.3 <u>MOVABLE CONTROL ASSEMBLIES</u>

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum **SHUTDOWN MARGIN** is maintained, and (3) the potential effects of a CEA ejection accident 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 criteria are met. A regulating or shutdown CEA is considered to be misaligned if it is more than 7.5 inches from any other CEA in its group, however, a shutdown CEA is also considered to be misaligned if it is withdrawn to less than 129 inches even if it is within 7.5 inches of all other CEAs in its group. For the purposes of the Technical Specifications, a dual assembly, connected to a single CEA drive mechanism, is considered to be a single CEA (e.g., dual shutdown CEAs connected to a single drive mechanism).

The **ACTION** statements applicable to an untrippable CEA and to a large misalignment ($\geq$ 15 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 an untrippable CEA, the loss of **SHUTDOWN MARGIN**. A CEA is considered untrippable when it is known that the CEA would not be insertable in response to a Reactor Protection System signal or is known to be immovable due to excessive friction or mechanical interference.

For small misalignments (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available **SHUTDOWN MARGIN**, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the **ACTION** statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA(s) to within their alignment requirements prior to initiating a reduction in **THERMAL POWER**. The one 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.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a single regulating or shutdown CEA. However, this misalignment would cause distortion of the core power distribution. The Reactor Protective System would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. The **ACTION** statement associated with a large CEA misalignment requires prompt action to realign the CEA to avoid excessive margin degradation. If the CEA is not realigned within the given time constraints, **ACTION** is specified which will preserve margin, including reductions in **THERMAL POWER**.

For a single CEA misalignment, the time allowance to realign the CEA is permitted for the following reasons:

1. The margin calculations which support the power distribution LCOs for DNBR are based on a steady-state F_r^T as specified in Technical Specification 3.2.3.

2. When the actual F_r^T is less than the Technical Specification value, additional margin exists.

3. This additional margin can be credited to offset the increase in F_r^T with time that will occur following a CEA misalignment due to xenon redistribution.

4. If an F_r^T measurement has not been taken recently (within 5 days), a pre-misaligned value of 1.70 is assumed and no time for realignment is permitted.

The requirement to reduce power level after the time limit is reached offsets the continuing increase in F_r^T that can occur due to xenon redistribution. A power reduction is not required below 50% power. Below 50% power there is sufficient conservatism in the DNB power distribution LCOs to completely offset any, or any additional, xenon redistribution effects.

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 brings 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.

There are five different operating modes for control of CEAs; Off, Manual Individual, Manual Group, Manual Sequential and Automatic. The Manual Sequential mode is applicable to only the regulating CEAs and the Automatic mode is disabled and not used for both regulating and shutdown CEAs.

OPERABILITY of the CEA position indicators is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. 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 **OPERABILITY** and the **ACTION** statements applicable to inoperable CEA position indicators permit continued operations when positions of CEAs with inoperable 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 LCOs are satisfied.

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 Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits.

Normally, the Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits. The setpoints for these alarms include allowances described in the COLR.

The control channel excore detectors, in conjunction with the Power Ratio Recorder Monitoring System, can also perform this function by continuously monitoring the **AXIAL SHAPE INDEX** and verifying that the **AXIAL SHAPE INDEX** is maintained within the allowable limits. In conjunction with the use of the Excore Monitoring System and in establishing the **AXIAL SHAPE INDEX** limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the **AZIMUTHAL POWER TILT** restrictions of Specification 3.2.4 are satisfied, 3) the **TOTAL PLANAR RADIAL PEAKING FACTOR** does not exceed the limits of Specification 3.2.2.

3/4.2.2, 3/4.2.3, and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND **AZIMUTHAL POWER TILT** - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, F^T_r, or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the **ACTION** statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An **AZIMUTHAL POWER TILT** > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The surveillance requirements for verifying that F^T_{xy}, F^T_r and T_q are within their limits provide assurance that the actual values of F^T_{xy}, F^T_r and T_q do not exceed the assumed values. Verifying F^T_{xy}, F^T_r after each fuel loading prior to exceeding 70% of **RATED THERMAL POWER** provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNB Specified Acceptable Fuel Design Limit (SAFDL) throughout each analyzed transient.

In addition to the DNB criterion, there are two other criteria which influence the DNB axial flux offset control limits. The second criterion is to ensure that the existing core power distribution at full power is less severe than the power distribution factored into the small-break LOCA analysis. This results in a limitation on the allowed negative **AXIAL SHAPE INDEX** value at full power. The third criterion is to maintain limitations on peak linear heat rate at low power levels resulting from Anticipated Operational Occurrences (AOOs). The DNB axial flux offset control limits are used to assure the LHR criteria for this condition because the linear heat rate LCO, for both ex-core and in-core monitoring, is set to maintain only the LOCA kw/ft requirements which are limiting at high power levels. At reduced power levels, the kw/ft requirements of certain AOOs (e.g., CEA withdrawal), tend to become more limiting than that for LOCA.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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% of its design pressure of 1000 psia during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18 x 10^6 lbs/hr at 100% **RATED THERMAL POWER**. 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 main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code.

In **MODE** 3, two main steam safety valves are required **OPERABLE** per steam generator. These valves will provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the Reactor Coolant System via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and **OPERABILITY** testing of the safety valves which can only be conducted when the RCS is at or above 500°F. It allows entry into **MODE** 3 with a minimum number of main steam safety valves **OPERABLE** so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

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:

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

where:

SP = reduced reactor trip setpoint in percent of **RATED THERMAL POWER**

V = maximum number of inoperable safety valves per steam line

106.5 = Power Level - High Trip Setpoint

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 Auxiliary Feedwater System

The **OPERABILITY** of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A delivered flow of 300 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the Shutdown Cooling System may be placed into operation.

Flow control valves were installed in the system in order to allow automatic flow initiation to a value selected by the Operator. Maximum flow to the steam generators from the motor driven AFW pump powered from the diesel is 300 gpm when feeding both generators (i.e., 150 gpm per leg maximum flow). The flow control valves installed in each leg supplied from the motor driven AFW pump shall be set at a flow setpoint not to exceed 150 gpm per leg. If the flow is only being directed to one steam generator, it is acceptable to deliver a maximum of 330 gpm because the flow error associated with the non-used loop is eliminated. These motor driven AFW pump capacity limits are imposed to prevent exceeding the emergency diesel generator load limit. If diesel generator loading is not a limiting concern, the delivered flow from the motor driven AFW pump may be increased up to a maximum of 575 gpm (motor HP limit vice diesel loading limit). These upper flow limits do not apply to the steam driven pumps.

Auxiliary Feedwater flow and response times are conservatively accounted for in the analyses of Design Basis Events. In Main Steam Line Break and Excess Load analyses where Auxiliary Feedwater flow would increase the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is minimized and Auxiliary Feedwater flow is maximized. In Feedline Break, Loss of Feedwater, and Loss of Non-Emergency AC Power Analyses, in which Auxiliary Feedwater flow would decrease the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is maximized and Auxiliary Feedwater flow is minimized.

At 10 minutes after an Auxiliary Feedwater Actuation Signal the operator is assumed to be available to increase or decrease auxiliary feedwater flow to that required by the existing plant conditions.

3/4.7.1.3 Condensate Storage Tank

The **OPERABILITY** of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at **HOT STANDBY** conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 Activity

The limitations on Secondary System specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 Main Steam Line Isolation Valves

The **OPERABILITY** of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The **OPERABILITY** of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses. The main steam isolation valves are surveilled to close in less than 5.2 seconds to ensure that under reverse steam flow conditions, the valves will close in less than the 6.0 seconds assumed in the accident analysis.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 80°F and 200 psig are based on steam generator secondary side limitations and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

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

3/4.7.4 SERVICE WATER SYSTEM

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

3/4.7.5 SALTWATER SYSTEM

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

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The **OPERABILITY** of the Control Room Emergency Ventilation 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 10 CFR Part 50, Appendix A, General Design Criteria 19.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The **OPERABILITY** of the ECCS Pump Room Exhaust Air Filtration System ensures that radioactive materials leaking from 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 effects on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 SNUBBERS

All safety related 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 non-safety 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.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during the previous inspection, the total population or category size, and the previous inspection interval.

Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and that decision shall be used as the basis upon which to determine the next inspection interval for that category. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

A snubber is considered inoperable if it fails to satisfy the acceptance criteria of the visual inspection. When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system. Operation may continue indefinitely if an engineering review and evaluation can document within 72 hours that the equipment connected to the snubber can continue to perform its required function(s) with the snubber inoperable. If the review and evaluation cannot justify that the supported equipment will perform its required function(s), the equipment must be declared inoperable and the applicable action requirements met.

The Specification allows inspection intervals to be compatible with a 24-month fuel cycle, up to and including an increase to every other refueling outage. To provide assurance of snubber functional reliability, a representative sample of the installed snubbers of each type* will be functionally tested during plant shutdowns at **REFUELING INTERVALS**. Observed failures of these sample snubbers shall require functional testing of additional units.

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* Small bore ($\leq$ 8") and large bore (> 8") hydraulic snubbers are examples of different types of snubbers.

The service life of a snubber is evaluated 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. The service life program is designed to uniquely reflect the conditions at Calvert Cliffs. The criteria for evaluating service life shall be determined, and documented, by the licensee. Records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

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.

3/4.7.10 WATERTIGHT DOORS

This specification is provided to ensure the protection of safety related equipment from the effects of water or steam escaping from ruptured pipes or components in adjoining rooms.

3/4.7.11 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 and fire hose stations. 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. Where a continuous fire watch is required in lieu of fire protection equipment and habitability due to heat or radiation is a concern, the fire watch should be stationed in a habitable area as close as possible to the inoperable equipment.

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. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.12 PENETRATION FIRE BARRIERS

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, a continuous fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration 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 volumes having direct access to the reactor vessel. A K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

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 accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

During **CORE ALTERATIONS** or movement of irradiated fuel within containment, a release of fission product radioactivity to the environment must be prevented. During **MODES** 1, 2, 3, and 4, this is accomplished by maintaining **CONTAINMENT INTEGRITY** as described in LCO 3.6.1. In other situations, the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment from the outside atmosphere.

The containment structure serves to contain fission product radioactivity which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR Part 100. Additionally, this structure provides radiation shielding from the fission products which may be present in the containment atmosphere following accident conditions.

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**.

The 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 identified by the annual land use census pursuant to Specification 3.12.2.

6.9.1.9 CORE OPERATING LIMITS REPORT

ore operating limits shall be established prior to each reload cle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.2.1
3.1.1.1
3.1.1.2
3.1.1.4
3.1.3.1
3.1.3.6
3.2.1
3.2.2.1
3.2.3
3.2.5
3.9.1

b. The analytical methods used to determine the core operating limits l be those previously reviewed and approved by the NRC; sp fically, those described in the fol owing documents:

(1) CENPD-199-P, Latest Approved Revision, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986

(2) CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II," December 1979

(3) CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," January 1980

(4) CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2," March 1980

(5) CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981

(6) Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

(7) CEN-348(B)-P, "Extended Statistical Combination of Uncertainties," January 1987

(8) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

(9) CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986

(10) CENPD-162-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

(11) CENPD-207-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"

(12) CENPD-206-P-A, Latest Approved Revision, "TORC Code, Verification and Simplified Modeling Methods"

(13) CENPD-225-P-A, Latest Approved Revision, "Fuel and Poison Rod Bowing"

(14) CENPD-266-P-A, Latest Approved Revision, "The ROCS and DIT Computer Code for Nuclear Design"

(15) CENPD-275-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"

(16) CENPD-382-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Erbium Burnable Absorbers"

(17) CENPD-139-P-A, Latest Approved Revision, "C-E Fuel Evaluation Model Topical Report"

(18) CEN-161-(B)-P-A, Latest Approved Revision, "Improvements to Fuel Evaluation Model"

(19) CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1989

(20) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

(21) CEN-372-P-A, Latest Approved Revision, "Fuel Rod Maximum Allowable Gas Pressure"

(22) Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"

(23) CENPD-132, Supplement 3-P-A, Latest Approved Revision, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"

(24) CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985

(25) CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985

(26) Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"

(27) CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977

(28) Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"

(29) CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977

(30) CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977

(31) Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"

(32) CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977

(33) Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"

(34) Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"

(35) Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"

(36) Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"

(37) Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"

(38) Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"

(39) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

(40) CENPD-188-A, Latest Approved Revision, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"

(41) The Full Core Power Distribution Monitoring System referenced in Specifications 3.1.3.1, 3.2.2.1, 3.2.3, and the BASES is described in the following documents:

(a) CENPD-153-P, Latest Approved Revision, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"

(b) CEN-199(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2," November 1979

(c) Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"

(d) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified f(each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Instrumentation, Specification 3.3.3.4.

d. Seismic Event Analysis, Specification 4.3.3.3.2.

e. Core Barrel Movement, Specification 3.4.11.

f. Fire Detection Instrumentation, Specification 3.3.3.7.

g. Fire Suppression Systems, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, 3.7.11.4, and 3.7.11.5.

h. Penetration Fire Barriers, Specification 3.7.12.

i. Steam Generator Tube Inspection Results, Specification 4.4.5.5.a and c.

j. Specific Activity of Primary Coolant, Specification 3.4.8.

k. Containment Structural Integrity, Specification 4.6.1.6.

l. Radioactive Effluents - Calculated Dose and Total Dose, Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3, and 3.11.4.

m. Radioactive Effluents - Liquid Radwaste, Gaseous Radwaste and Ventilation Exhaust Treatment Systems Discharges, Specifications 3.11.1.3 and 3.11.2.4.

n. Radiological Environmental Monitoring Program, Specification 3.12.1.

o. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6).

p. Overpressure Protection Systems, Specification 3.4.9.3.

q. Hydrogen Analyzers, Specification 3.6.5.1.

r. Post-Accident Instrumentation, Specification 3.3.3.6.

6.10 RECORD RETENTION

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

a. Records and logs of facility 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 reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

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

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

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

a. Records and drawing changes reflecting facility 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 facility radiation and contamination surveys.

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

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

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

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

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

i. Records of Quality Assurance activities identified in the NRC approved QA Manual as lifetime records.

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 POSRC, the Procedure Review Committee, and the OSSRC.

l. Records of the service lives of all safety related snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2):

a. A high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Special or Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A high radiation area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a, above, and in addition locked barricades shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained by the Supervisor - Radiation Control Operations and the Operations Shift Supervisor on duty under their separate administrative control.

6.13 SYSTEM INTEGRITY

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

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.14 IODINE MONITORING

The licensee shall implement 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:

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

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* It is acceptable if the licensee maintains details of the program in plant operation manuals (e.g., chemistry procedures, training instructions, maintenance procedures, ERPIPs).

6.15 POSTACCIDENT SAMPLING

The licensee shall establish, implement and maintain 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:

a. Training of personnel,

b. Procedures for sampling and analysis, and

c. Provisions for maintenance of sampling and analysis equipment.

6.16 PROCESS CONTROL PROGRAM (PCP)

6.16.1 The **PCP** shall be approved by the Commission prior to implementation.

6.16.2 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. An evaluation supporting the premise that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   2. A reference to the date and the POSRC meeting number in which the change(s) was reviewed and found acceptable to the POSRC.

b. Shall become effective upon review by the POSRC and approval of the Plant General Manager.

6.17 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.17.1 The **ODCM** shall be approved by the Commission prior to implementation.

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* It is acceptable if the licensee maintains details of the program in plant operation manuals (e.g., chemistry procedures, training instructions, maintenance procedures, ERPIPs).

6.17.2 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. Sufficient information to support the rationale for the change. Information submitted should consist of a package of those pages of the **ODCM** to be changed with each page numbered and provided with a change number and/or change date 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. Documentation of the fact that the change has been reviewed and found acceptable by the POSRC.

b. Shall become effective upon review by the POSRC and approval of the Plant General Manager.

6.18 <u>MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS</u>

6.18.1 Licensee initiated major changes to the Radioactive Waste Systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the modification to the waste system is completed. The discussion of each change shall contain:

a. A description of the equipment, components and processes involved.

b. Documentation of the fact that the change including the safety analysis was reviewed and found acceptable by the POSRC.

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ADMINISTRATIVE CONTROLS

CONTROLLED LEAKAGE

1.9 **CONTROLLED LEAKAGE** shall be the water flow from the reactor coolant pump seals.

CORE ALTERATION

1.10 **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.

CORE OPERATING LIMITS REPORT

1.11 The **CORE OPERATING LIMITS REPORT** is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

1.12 **DOSE EQUIVALENT I-131** shall be that concentration of I-131 (μCi/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

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.17 **IDENTIFIED LEAKAGE** shall be:

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

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

c. Reactor Coolant System leakage through a steam generator to the Secondary System.

MEMBER(S) OF THE PUBLIC

1.18 **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 utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.19 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

1.20 A system, subsystem, train, component or device shall be **OPERABLE** or have **OPERABILITY** when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other required 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

1.21 An **OPERATIONAL MODE** shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.22 **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 13.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.23 **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)

1.24 The **PROCESS CONTROL PROGRAM** shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State and local regulations governing the disposal of the radioactive waste.

PURGE - PURGING

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

RATED THERMAL POWER

1.26 **RATED THERMAL POWER** shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 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

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

SHUTDOWN MARGIN

1.29 **SHUTDOWN MARGIN** shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length 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

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

SOLIDIFICATION

1.31 **SOLIDIFICATION** shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

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.

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

1.35 The **TOTAL INTEGRATED RADIAL PEAKING FACTOR** is the ratio of the peak pin power to the average pin power in an unrodded core.

TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

1.36 The **TOTAL PLANAR RADIAL PEAKING FACTOR** is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes.

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.38 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

1.39 A **VENTILATION EXHAUST TREATMENT SYSTEM** is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. 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

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of **THERMAL POWER**, pressurizer pressure, and highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1-1.

APPLICABILITY: **MODES** 1 and 2.

ACTION: Whenever the point defined by the combination of the highest operating loop cold leg temperature and **THERMAL POWER** has exceeded the appropriate pressurizer pressure line, be in **HOT STANDBY** within 1 hour.

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.

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.

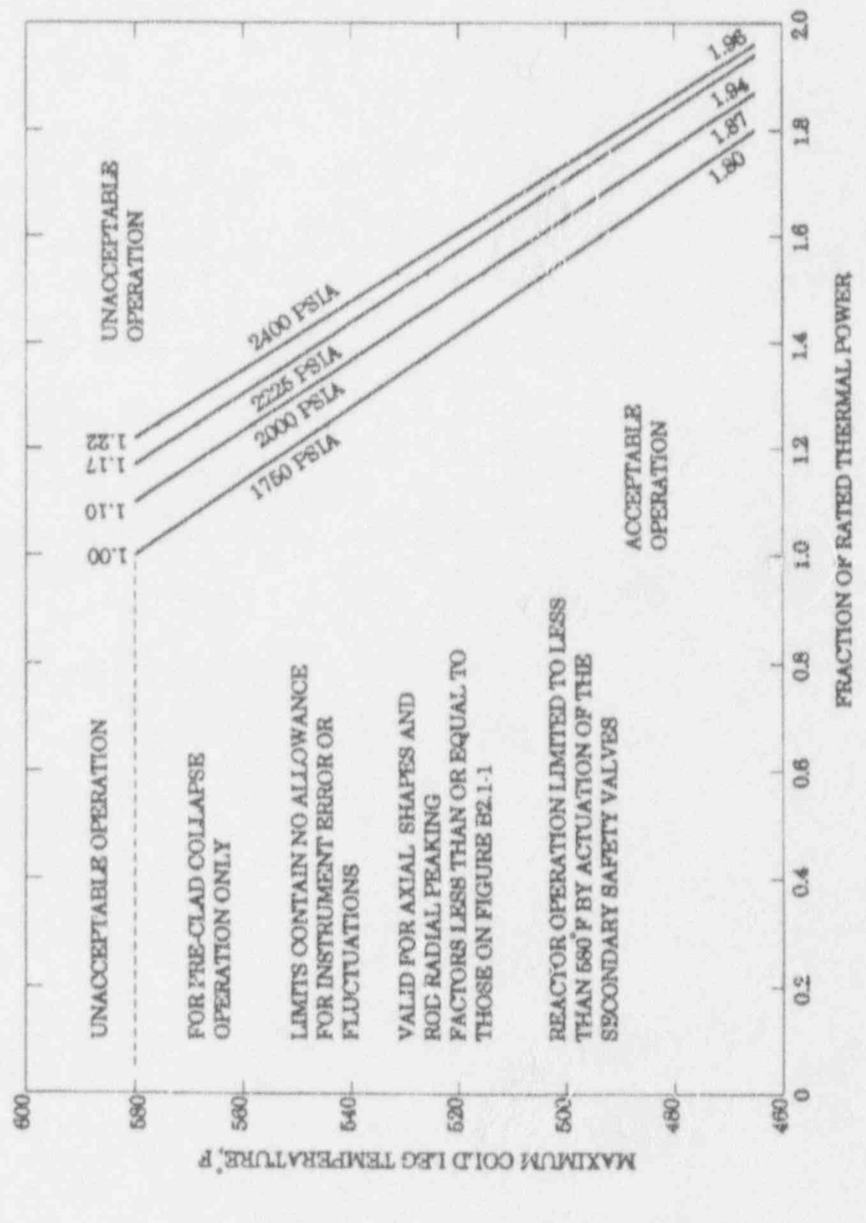

FIGURE 2.1-1

REACTOR CORE THERMAL MARGIN SAFETY LIMIT

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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 3.3.1.1 until the channel is restored to **OPERABLE** status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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. Power Level - High	≤ 10% above **THERMAL POWER**, with a minimum setpoint of 30% of **RATED THERMAL POWER**, and a maximum of ≤ 107.0% of **RATED THERMAL POWER**.	≤ 10% above **THERMAL POWER**, and a minimum setpoint of 30% of **RATED THERMAL POWER** and a maximum of ≤ 107.0% of **RATED THERMAL POWER**.
3. Reactor Coolant Flow - Low (1)	≥ 95% of design reactor coolant flow*	≥ 95% of design reactor coolant flow*
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 4 psig	≤ 4 psig
6. Steam Generator Pressure - Low (2)	≥ 685 psia	≥ 685 psia
7. Steam Generator Water Level - Low	≥ 10 inches below top of feed ring	≥ 10 inches below top of feed ring

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Axial flux offset [3]	Trip setpoint adjusted to not exceed the limits provided in the COLR	Trip setpoint adjusted to not exceed the limits provided in the COLR
9. Thermal Margin/Low Pressure [1]		
a. Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limits provided in the COLR	Trip setpoint adjusted to be not less than the larger of (1) 1875 psia, or (2) the limits provided in the COLR
b. Steam Generator Pressure Difference - High [1]	$\leq$ 135 psid	$\leq$ 135 psid
10. Loss of Load	NA	NA
11. Rate of Change of Power - High [4]	$\leq$ 2.6 decades per minute	$\leq$ 2.6 decades per minute

TABLE 2.2-1 (Continued)

TABLE NOTATION

* See Specification 3.2.5, "DNB Parameters," for the design reactor coolant flow.

(1) Trip may be bypassed below 10^{-4}% OF **RATED THERMAL POWER**; bypass shall be automatically removed when **THERMAL POWER** is $\geq 10^{-4}$% of **RATED THERMAL POWER**.

(2) Trip may be manually bypassed below 785 psia; bypass shall be automatically removed at or above 785 psia.

(3) Trip may be bypassed below 15% of **RATED THERMAL POWER**; bypass shall be automatically removed when **THERMAL POWER** is $\geq$ 15% of **RATED THERMAL POWER**.

(4) Trip may be bypassed below 10^{-4}% and above 12% of **RATED THERMAL POWER**.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which could result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 22.0 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by 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.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and, therefore, **THERMAL POWER** and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the DNB SAFDL of 1.15 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This DNB SAFDL assures with at least a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curves of Figure 2.1-1 show conservative loci of points of **THERMAL POWER**, Reactor Coolant System pressure and maximum cold leg temperature for which the DNB SAFDL is not violated for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. The area of safe operation is below and to the left of these lines. Reactor operation at **THERMAL POWER** levels higher than 110% of **RATED THERMAL POWER** is prohibited by the high power level trip setpoint specified in Table 2.2-1.

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Reactor Protective System in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and **THERMAL POWER** level that would result in a DNBR of less than 1.15, in conjunction with the ESCU methodology, and preclude the existence of flow instabilities.

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 pressure vessel and pressurizer are designed to Section III, 1967 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves, and fittings are designed to ANSI B 31.7, Class I, 1968 Edition, which permits a maximum transient pressure of 110% (2750 psia) of component 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 LIMITING SAFETY SYSTEM SETTINGS

BASES

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 parameter. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their safety limits. 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 the trip setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

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

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 10% above the indicated **THERMAL POWER** level. Operator action is required to increase the trip setpoint as **THERMAL POWER** is increased. The trip setpoint is automatically decreased as **THERMAL POWER** decreases. The trip setpoint has a maximum value of 107.0% of **RATED THERMAL POWER** and a minimum setpoint of 30% of **RATED THERMAL POWER**. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state **THERMAL POWER** level at which a trip would be actuated is 110% of **RATED THERMAL POWER**, which is the value used in the safety analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. The low-flow trip setpoints and Allowable Values have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above the DNB SAFDL of 1.15, in conjunction with the ESCU methodology, under normal operation and expected transients.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to, or at least concurrently with, a safety injection.

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 setting of 685 psia is sufficiently below the full-load operating point of 850 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 setting was used with an uncertainty factor of $\pm$ 85 psi which was based on the Main Steam Line Break event inside containment.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the pressure of the Reactor Coolant System will not exceed its Safety Limit. The specified setpoint in combination with the Auxiliary Feedwater Actuation System ensures that sufficient water inventory exists in both steam generators to remove decay heat following a loss of main feedwater flow event.

Axial Flux Offset

The axial flux offset trip is provided to ensure that excessive axial peaking will not cause fuel damage. The axial flux offset is determined from the axially split excore detectors. The trip setpoints ensure that neither a DNBR of less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology, nor a peak linear heat rate which corresponds to the temperature for fuel centerline melting will exist as a consequence of axial power maldistributions. These trip setpoints were derived from an analysis of many axial power shapes with allowances for instrumentation inaccuracies and the uncertainty associated with the excore-to-incore axial flux offset relationship.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNB SAFDL of 1.15, in conjunction with ESCU methodology.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, and reactor inlet temperature. The minimum value of reactor coolant flow rate, the maximum **AZIMUTHAL POWER TILT**, and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints include allowances for equipment response time, measurement uncertainties, processing error and a further allowance of 40 psia to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF)

The ASGTPTF utilizes steam generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those Anticipated Operational Occurrences associated with Secondary System malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single Main Steam Isolation Valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

Loss of Load

A Loss of Load trip causes a direct reactor trip when operating above 15% of **RATED THERMAL POWER**. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability is required to enhance overall plant equipment service life and reliability.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during **STARTUP** operations and its use serves as a backup to the administratively enforced **STARTUP** rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the accident 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.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The **SHUTDOWN MARGIN** shall be within the limit provided in the COLR.

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

ACTION: With the SHUTDOWN MARGIN outside the limit provided in the COLR, immediately initiate and continue boration at ≥ 40 gpm of 2300 ppm boric acid solution or equivalent until the required **SHUTDOWN MARGIN** is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The **SHUTDOWN MARGIN** shall be determined to be within the limit provided in the COLR:

a. Within one 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 untrippable, the above required **SHUTDOWN MARGIN** shall be increased by an amount at least equal to the withdrawn worth of the untrippable CEA(s).

b. When in **MODES** 1 or 2 with K_{eff} ≥ 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.

---

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

c. When in **MODE** 2 with $K_{eff} < 1.0$, 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.

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 **MODES** 3 or 4, 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.

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 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The **SHUTDOWN MARGIN** shall be within the limit provided in the COLR, and when pressurizer level is less than 90 inches from bottom of the pressurizer, all sources of non-borated water shall be $\leq$ 88 gpm.

APPLICABILITY: **MODE** 5.

ACTION:

a. With the **SHUTDOWN MARGIN** outside the limit provided in the COLR, immediately initiate and continue boration at $\geq$ 40 gpm of 2300 ppm boric acid solution or equivalent until the required **SHUTDOWN MARGIN** is restored.

b. With the pressurizer drained to < 90 inches and all sources of non-borated water > 88 gpm, immediately suspend all operations involving positive reactivity changes while the **SHUTDOWN MARGIN** is increased to compensate for the additional sources of non-borated water or reduce the sources of non-borated water to $\leq$ 88 gpm.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The **SHUTDOWN MARGIN** shall be determined to be within the limit provided in the COLR:

a. Within one 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 untrippable, the above required **SHUTDOWN MARGIN** shall be increased by an amount at least equal to the withdrawn worth of the 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,

SURVEILLANCE REQUIREMENTS (Continued)

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

4.1.1.2.2 With the pressurizer drained to < 90 inches determine:

a. Within one hour and every 12 hours thereafter that the level in the Reactor Coolant System is above the bottom of the hot leg nozzles, and

b. Within one hour and every 12 hours thereafter that the sources of non-borated water are ≤ 88 gpm or the **SHUTDOWN MARGIN** has been increased to compensate for the additional non-borated water sources.

3/4.1.1 BORATION CONTROL

Boron Dilution

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the Reactor Coolant System shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: All **MODES**.

ACTION: With the flow rate of reactor coolant through the Reactor Coolant System < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the Reactor Coolant System shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm through the Reactor Coolant System.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.1 BORATION CONTROL

Moderator Temperature Coefficient

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the limits provided in the COLR, and

b. Less positive than the limit line of Figure 3.1.1-1.

APPLICABILITY: **MODES** 1 and 2 with $K_{eff} > 1.0$#.

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

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 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.

\# See Special Test Exception 3.10.2.

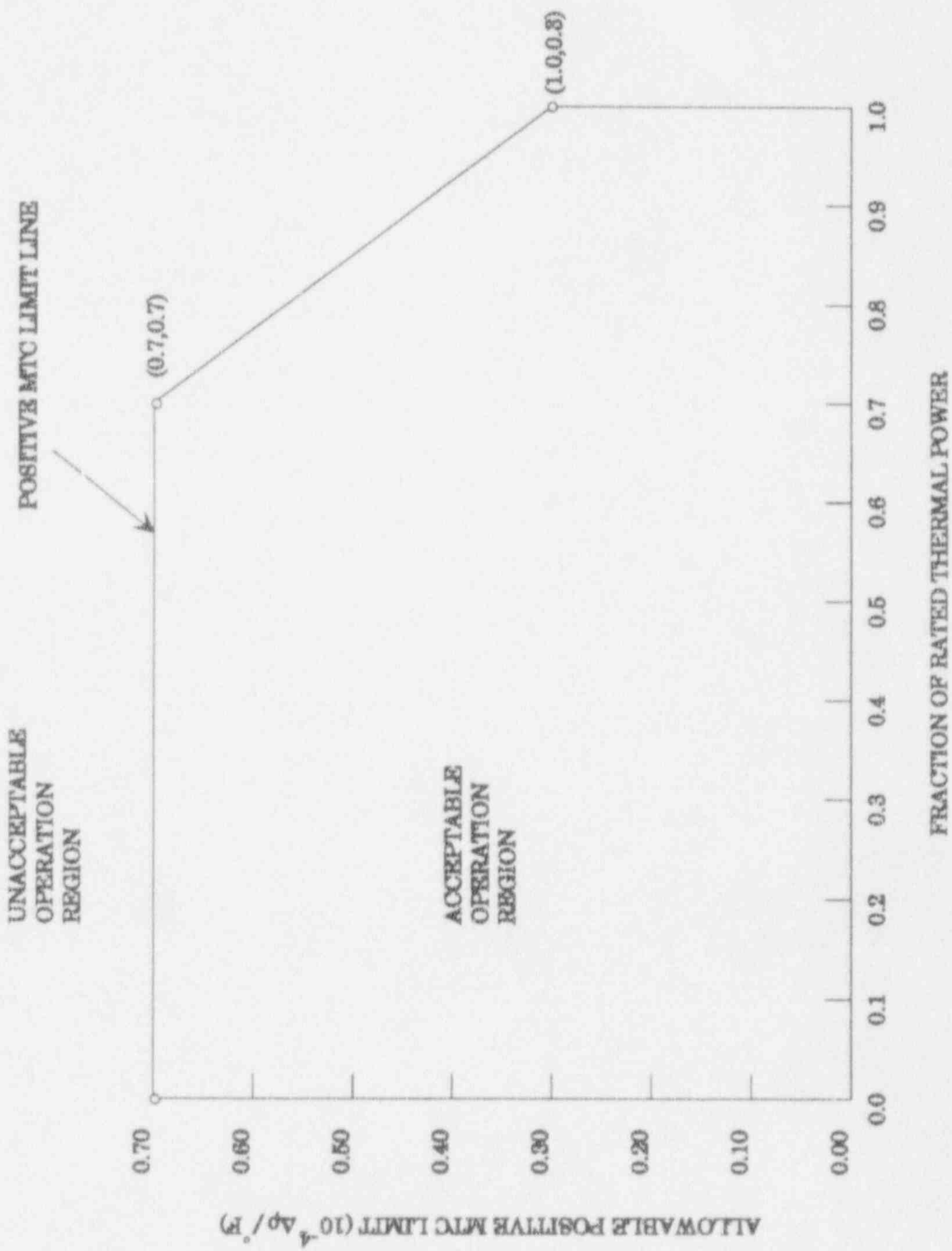

FIGURE 3.1.1-1

FRACTION OF RATED THERMAL POWER
VS. ALLOWABLE POSITIVE MTC LIMIT ($10^{-4}\Delta\rho/F°$)

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 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** above 90% of **RATED THERMAL POWER**, within 7 EFPD after initially reaching an equilibrium condition at or above 90% of **RATED THERMAL POWER** after each fuel loading.

c. At any **THERMAL POWER**, within 7 EFPD of reaching a **RATED THERMAL POWER** equilibrium boron concentration of 300 ppm.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

Minimum Temperature For Criticality

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 515°F when the reactor is critical.

APPLICABILITY: **MODES** 1 and 2 with $K_{eff} \geq 1.0$.

ACTION: With a Reactor Coolant System operating loop temperature (T_{avg}) < 515°F, restore T_{avg} to within its limit within 15 minutes or be in **HOT STANDBY** within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be $\geq$ 515°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_{avg} is less than 525°F.

3/4.1.2 BORATION SYSTEMS

Flow Paths - Shutdown

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. A flow path from the boric acid storage tank via either a boric acid pump or a gravity feed connection and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is **OPERABLE**, or

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

APPLICABILITY: **MODES** 5 and 6.

ACTION: With none of the above flow paths **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one injection path is restored to **OPERABLE** status.

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 3.1.2-1 when a flow path from the concentrated boric acid 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.

* At 305°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 305°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Flow Paths - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be **OPERABLE**:

a. Two flow paths from the boric acid storage tanks required to be **OPERABLE** pursuant to Specifications 3.1.2.8 and 3.1.2.9 via either a boric acid pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank 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 that required by Specification 3.1.1.1 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 concentrated boric acid tanks is above the temperature limit line shown on Figure 3.1.2-1.

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.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per **REFUELING INTERVAL** by verifying on a SIAS test signal that:

(1) each automatic valve in the flow path actuates to its correct position, and

(2) each boric acid pump starts.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Charging Pump - 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 bus.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With no charging pump or high pressure safety injection pump **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one of the required pumps is restored to **OPERABLE** status.

SURVEILLANCE REQUIREMENTS

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

* At 305°F and less, the required **OPERABLE** HPSI pump shall be in pull-to-lock and will not start automatically. At 305°F and less, HPSI pump use will be conducted in accordance with Technical Specification 3.4.9.3.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Charging Pumps - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two 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 that required by Specification 3.1.1.1 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 At least two charging pumps shall be demonstrated **OPERABLE**:

a. At least once per **REFUELING INTERVAL** by verifying that each charging pump starts automatically upon receipt of a Safety Injection Actuation Test Signal.

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

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* Above 80% **RATED THERMAL POWER** the two **OPERABLE** charging pumps shall have independent power supplies.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

Boric Acid Pumps - Shutdown

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid 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 above, is **OPERABLE**.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With no boric acid 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 until at least one boric acid pump is restored to **OPERABLE** status.

SURVEILLANCE REQUIREMENTS

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

Boric Acid Pumps - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.6

a. The boric acid pump(s) in the boron injection flow path(s) required to be **OPERABLE** pursuant to Specification 3.1.2.2.a shall be **OPERABLE** and capable of being powered from an **OPERABLE** emergency bus.

AND,

When in **MODE** 1 > 80% of **RATED THERMAL POWER**

b. The boric acid pump(s) in the boron injection flow path(s) required to be **OPERABLE** pursuant to Specification 3.1.2.8.a shall be **OPERABLE** and capable of being powered from an **OPERABLE** emergency bus.

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

ACTION: With one boric acid pump required for the boron injection flow path(s) pursuant to either Specification 3.1.2.2.a or 3.1.2.8.a inoperable, restore the boric acid 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 that required by Specification 3.1.1.1 at 200°F; restore the above required boric acid 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 No additional Surveillance Requirements other than those required by Specifications 4.0.5 and 4.1.2.2.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.2 BORATION SYSTEMS

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 storage tank and one associated heat tracing circuit with the tank contents in accordance with Figure 3.1.2-1.

b. The refueling water tank with:

1. A minimum contained borated water volume of 9,844 gallons,

2. A minimum boron concentration of 2300 ppm, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: **MODES** 5 and 6.

ACTION: With no borated water sources **OPERABLE**, suspend all operations involving **CORE ALTERATIONS** or positive reactivity changes until at least one borated water source is restored to **OPERABLE** status.

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 storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is < 35°F.

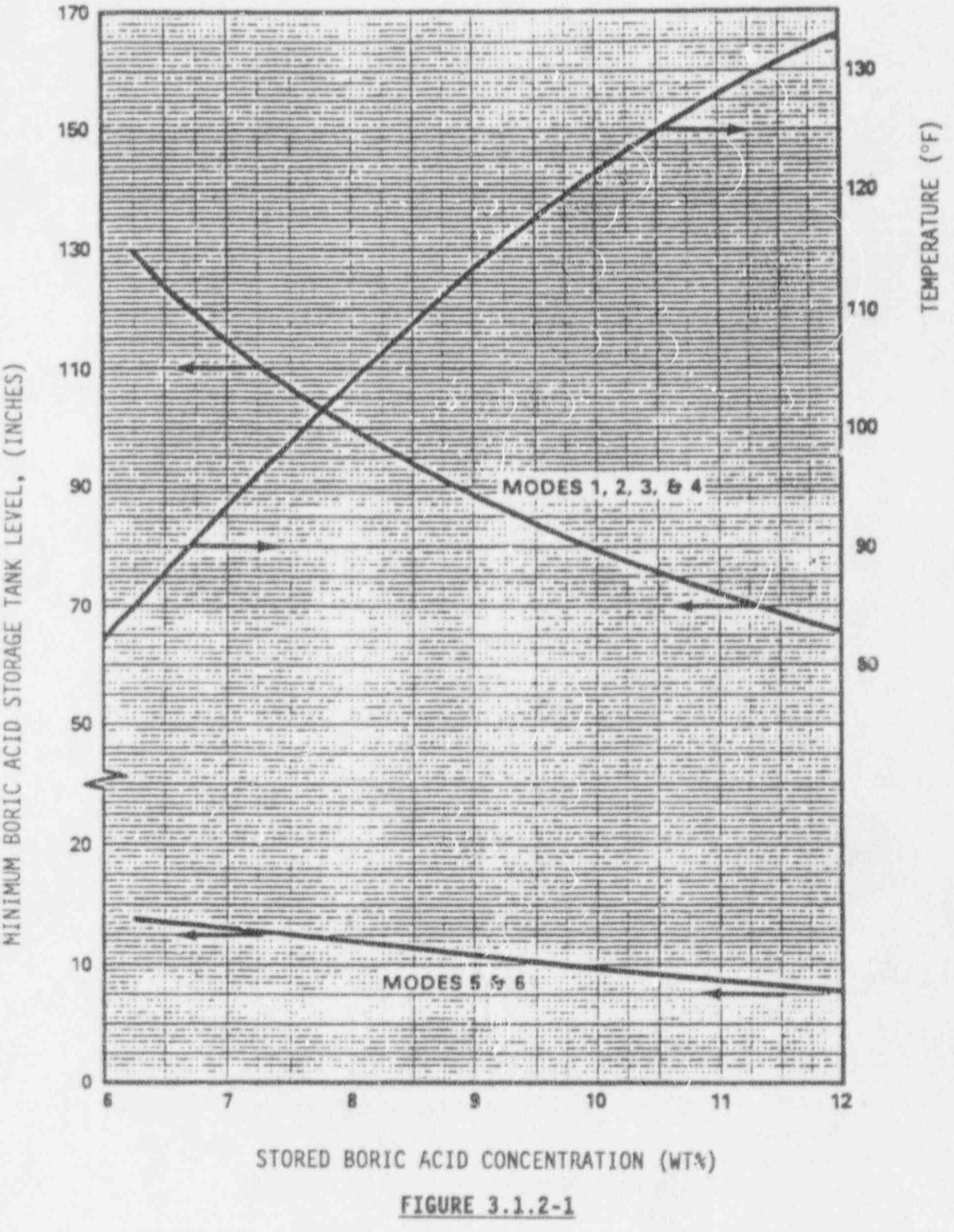

FIGURE 3.1.2-1

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

3/4.1.2 BORATION SYSTEMS

Charging Pump ECCS Subsystem

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, the following equipment shall be **OPERABLE**:

a. Boric Acid Storage Tank 22 and its associated heat tracing circuit shall be **OPERABLE** per Specification 3.1.2.9.a and the boron injection flow path via Boric Acid Pump 22 from Boric Acid Storage Tank 22 shall be **OPERABLE** per Specification 3.1.2.2.a and Specification 3.1.2.6.

AND,

One of the following:

b. The boron injection flow path from Boric Acid Storage Tank 22 via a gravity feed connection shall be **OPERABLE** per Specification 3.1.2.2.a, or,

Boric Acid Storage Tank 21 and its associated heat tracing circuit shall be **OPERABLE** per Specification 3.1.2.9.a and the boron injection flow path from Boric Acid Storage Tank 21 via a gravity feed connection shall be **OPERABLE** per Specification 3.1.2.2.a.

APPLICABILITY: **MODE** 1 > 80% of **RATED THERMAL POWER.**

ACTION: With only one of the required combinations of borated water sources and flow paths **OPERABLE**, restore two required combinations of borated water sources and flow paths to **OPERABLE** status within 72 hours or reduce power to less than 80% of **RATED THERMAL POWER** within the next 6 hours and comply with Specifications 3.1.2.2, 3.1.2.6, and 3.1.2.9 as applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.8 No additional Surveillance Requirements other than those required by Specifications 4.0.5, 4.1.2.2, and 4.1.2.9.

3/4.1.2 BORATION SYSTEMS

Borated Water Sources - Operating

LIMITING CONDITION FOR OPERATION

3.1.2.9 At least two of the following three borated water sources shall be **OPERABLE**:

a. Two boric acid storage tank(s) and one associated heat tracing circuit per tank with the contents of the tanks in accordance with Figure 3.1.2-1 and the boron concentration limited to $\leq$ 8%, and

b. The refueling water tank with:

1. A minimum contained borated water volume of 400,000 gallons,
2. A boron concentration of between 2300 and 2700 ppm,
3. A minimum solution temperature of 40°F, and
4. A maximum solution temperature of 100°F in **MODE** 1.

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

ACTION: With only one borated water source **OPERABLE**, restore at least two borated water sources 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 that required by Specification 3.1.1.1 at 200°F; restore at least two borated water sources to **OPERABLE** status within the next 7 days or be in **COLD SHUTDOWN** within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.9 At least two borated water sources shall be demonstrated **OPERABLE**:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume in each water source, and

3. Verifying the boric acid storage tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is $< 40°F$.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA Position

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Motion Inhibit and all shutdown and regulating CEAs shall be **OPERABLE** with each CEA of a given group positioned within 7.5 inches (indicated position) of all other CEAs in its group.

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

ACTION:

a. With one or more CEAs (regulating or shutdown) inoperable due to being untrippable, be in at least **HOT STANDBY** within 6 hours.

b. With the CEA Motion Inhibit inoperable, within 6 hours either:

   1. Restore the CEA Motion Inhibit to **OPERABLE** status, or

   2. Fully withdraw all CEAs in groups 3 and 4 and withdraw the CEAs in group 5 to less than 5% insertion, or

   3. Be in at least **HOT STANDBY**.

c. With one regulating 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 for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year.

d. With one CEA (regulating or shutdown) inoperable due to causes other than addressed by **ACTION** a above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in CEA group 5, operation in **MODES** 1 and 2 may continue.

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* See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

e. With one CEA (regulating or shutdown) misaligned from any other CEA in its group by less than 15 inches, operation in **MODES** 1 and 2 may continue, provided that the misaligned CEA is restored to within its specified alignment requirements within one hour, otherwise implement Action g.

f. With one CEA (regulating or shutdown) misaligned from any other CEA in its group by 15 inches or more, operation in **MODES** 1 and 2 may continue, provided that the misaligned CEA is restored to within its specified alignment requirements within the time allowance determined by the full core power distribution monitoring system or, if the full core power distribution monitoring system time allowance is unavailable, the allowable time shall be that provided in the COLR, otherwise implement Action g. If the COLR allowable time is used, the pre-misaligned F_r^T value used to determine the allowable time to realign the CEA shall be the latest measurement taken within 5 days prior to the CEA misalignment. If no measurements were taken within 5 days prior to the misalignment, immediately implement Action g.

LIMITING CONDITION FOR OPERATION (Continued)

g. With one CEA (regulating or shutdown) not within its specified alignment requirements at the conclusion of the time allowance permitted by Actions e, f or h, immediately start to implement the following actions:

   1. If the **THERMAL POWER** level prior to the misalignment was greater than 50% of **RATED THERMAL POWER**, **THERMAL POWER** shall be reduced to less than the greater of:

      a) 50% of **RATED THERMAL POWER**

      b) 75% of the **THERMAL POWER** level prior to the misalignment

   within one hour after exceeding the time allowance permitted by Actions e, f or h.

   2. If the **THERMAL POWER** level prior to the misalignment was $\leq$ 50% of **RATED THERMAL POWER**, maintain **THERMAL POWER** no higher than the value prior to the misalignment.

   If negative reactivity insertion is required to reduce **THERMAL POWER**, boration shall be used. Within one hour after establishing the appropriate **THERMAL POWER** as required above, either:

   1. Restore the CEA to within its specified alignment requirements, or

   2. Declare the CEA inoperable. After declaring the CEA inoperable, **POWER OPERATION** may continue for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year provided that within one hour after declaring the CEA inoperable, the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 inches of the inoperable CEA while:

      a) maintaining the allowable CEA sequence and CEA group insertion limits located in the COLR; for a regulating CEA, and with the **THERMAL POWER** level restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

      b) maintaining the remainder of the CEAs in a shutdown group withdrawn to at least 129 inches.

LIMITING CONDITION FOR OPERATION (Continued)

h. With more than one CEA (regulating or shutdown) misaligned and each misaligned CEA is within 15 inches of any other CEA in its group (indicated position) restore the misaligned CEAs to within their specified alignment requirements within one hour, otherwise immediately declare the misaligned CEAs inoperable and implement Action i. If only one CEA (regulating or shutdown) remains misaligned at the end of one hour, implement Action g.

i. With more than one CEA (regulating or shutdown) inoperable or with more than one CEA (regulating or shutdown) misaligned and any one or more of the misaligned CEAs is 15 inches (indicated position) or more from any other CEA in its group, be in at least **HOT STANDBY** within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Motion Inhibit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each CEA not fully inserted shall be determined to be **OPERABLE** by inserting it at least 7.5 inches at least once per 31 days. For the purposes of performing the CEA operability test, if the CEA has an inoperable position indication channel, the alternate indication system (pulse counter or voltage dividing network) will be used to monitor position. If a direct position indication (full out reed switch or voltage dividing network) cannot be restored within ten minutes from the commencement of CEA motion, or CEA withdrawal exceeds the surveillance testing insertion by > 7.5 inches, the position of the CEA shall be assumed to have been > 15 inches from its group at the commencement of CEA motion.

4.1.3.1.3 The CEA Motion Inhibit shall be demonstrated **OPERABLE** at least once per 31 days by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit also prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

Position Indicator Channels

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least two of the following three CEA position indicator channels shall be **OPERABLE** for each shutdown and regulating CEA:

a. CEA voltage divider reed switch position indicator channel, capable of determining the absolute CEA position within $\pm$ 1.75 inches;

b. CEA "Full Out" or "Full In" reed switch position indicator channel, only if the CEA is fully withdrawn or fully inserted, as verified by actuation of the applicable position indicator; and

c. CEA pulse counting position indicator channel.

APPLICABILITY: **MODES** 1 and 2.

ACTION:

a. With a maximum of one CEA per group having its voltage divider reed switch position indicator channel or its pulse counting position indicator channel inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, either:

1. Within 6 hours

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

b) Be in at least **HOT STANDBY**, or

c) Reduce **THERMAL POWER** to $\leq$ 70% of **RATED THERMAL POWER**; if negative reactivity insertion is required to reduce **THERMAL POWER**, boration shall be used. Operation at or below this reduced **THERMAL POWER** level may continue provided that within the next 4 hours either:

1) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining

the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the **THERMAL POWER** level may be returned to a level consistent with all other applicable specifications and operation may continue per Specification 3.1.3.3 above; or

2) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and **THERMAL POWER** level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6, and may continue per Specification 3.1.3.3 above.

or,

2. If the failure existed before entry into **MODE** 2 or occurs prior to an "all CEAs out" configuration, the CEA group(s) with inoperable position indicator channel must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator. These actions must be completed within 10 hours of entry into **MODE** 2 and prior to exceeding 70% of **RATED THERMAL POWER**. The provisions of Specification 3.0.4 are not applicable. Once these actions are completed, operation may continue per Specification 3.1.3.3 above.

b. With more than one CEA per group having its CEA pulse counting position indicator channel and either (1) the "Full Out" or "Full In" position indicator, or (2) the voltage divider position indicator channel inoperable, operation in **MODES** 1 and

LIMITING CONDITION FOR OPERATION (Continued)

2 may continue for up to 24 hours provided that for the affected CEAs, either:

1. The CEA voltage divider reed switch position indicator channels are **OPERABLE**, or

2. The CEA "Full Out" or "Full In" reed switch position indicator channels are **OPERABLE**, with the CEA fully withdrawn or fully inserted as verified by actuation of the applicable position indicator.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each required CEA position indication channel shall be determined to be **OPERABLE** by determining CEA positions as follows at least once per 12 hours, by:

a. Verifying the CEA pulse counting position indicator channels and the CEA voltage divider reed switch position indicator channels agree within 4.5 inches, or

b. Verifying the CEA pulse counting position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches, or

c. Verifying the CEA voltage divider reed switch position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches.

4.1.3.3.2 During time intervals when the deviation circuit is inoperable, the above verification of required CEA position indicator channels shall be made at least once per 4 hours.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA Drop Time

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: **MODES** 1 and 2.

ACTION: 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.

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 of the reactor vessel head,

b. For specifically affected individual 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 **REFUELING INTERVAL**.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

Shutdown CEA Insertion Limit

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to at least 129.0 inches.

APPLICABILITY: **MODES** 1 and 2 with $K_{eff} \geq 1.0^{*}$.

ACTION: With one or more shutdown CEA(s) withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 129.0 inches, consider the CEA(s) misaligned, and immediately apply Specification 3.1.3.1; Action e, f, h or i, as appropriate.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to at least 129.0 inches:

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.

3/4.1 **REACTIVITY CONTROL SYSTEMS**

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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 provided in the COLR (regulating CEAs are considered to be fully withdrawn when withdrawn to at least 129.0 inches) with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. $\leq$ 4 hours per 24 hour interval,

b. $\leq$ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. $\leq$ 14 Effective Full Power Days per calendar year.

APPLICABILITY: **MODES** 1* and 2 with $K_{eff} \geq 1.0$*.

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, within two 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 limits provided in the COLR.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 4 hours per 24 hour interval, except during operations pursuant to the provisions of **ACTION** items c. and e. of Specification 3.1.3.1, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits provided in the COLR are not exceeded, or

2. Any subsequent increase in **THERMAL POWER** is restricted to $\leq$ 5% of **RATED THERMAL POWER** per hour.

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* See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of **ACTION** items c. and e. of Specification 3.1.3.1, 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 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 Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 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 provided in the COLR.

APPLICABILITY: **MODE** 1.

ACTION: With the linear heat rate exceeding the limits provided in the COLR, as indicated by four or more coincident incore channels or by the **AXIAL SHAPE INDEX** outside of the power dependent control limits provided in the COLR, 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 one 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 by continuously monitoring the core power distribution with either the Excore Detector Monitoring System or with the Incore Detector Monitoring System.

4.2.1.3 Excore Detector Monitoring System - The Excore Detector Monitoring System may be used for monitoring the core power distribution* by:

a. Verifying at least once per 12 hours that the CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. Verifying at least once per 31 days that the **AXIAL SHAPE INDEX** alarm setpoints are adjusted to within the limits provided in the COLR.

* See Surveillance 4.2.2.1.2.d

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the **AXIAL SHAPE INDEX** is maintained within the limits provided in the COLR, where 100 percent of the allowable power is the maximum allowable fraction of **RATED THERMAL POWER** as determined by the F^T_{xy} limit provided in the COLR.

4.2.1.4 Incore Detector Monitoring System - The Incore Detector Monitoring System may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days* of accumulated operation in **MODE** 1.

b. Have their alarm setpoint adjusted to protect the linear heat rate limit provided in the COLR when the uncertainty factors provided in the COLR** are appropriately included in the setting of these alarms.

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* For Unit 2 Cycle 10 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, this surveillance shall be performed at least once per 15 days of accumulated operation in **MODE** 1.

** For Unit 2 Cycle 10 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, the measurement-calculational uncertainty factor on linear heat rate shall be increased by 1% (from 1.062 to 1.072) prior to comparison with the Technical Specification limit.

3/4.2 **POWER DISTRIBUTION LIMITS**

3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2.1 The calculated value of F^T_{xy} shall be within the limit provided in the COLR.**

APPLICABILITY: **MODE 1***.

ACTION: With F^T_{xy} outside the limit provided in the COLR, within 6 hours either:

a. Withdraw and maintain the CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce **THERMAL POWER** as follows:

   1. Reduce **THERMAL POWER** to bring the combination of **THERMAL POWER** and F^T_{xy} within the limits provided in the COLR, or

   2. Reduce **THERMAL POWER** to less than or equal to the limit established by the full core power distribution monitoring system as a function of F^T_{xy}; or

b. Be in at least **HOT STANDBY**.

SURVEILLANCE REQUIREMENTS

4.2.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.1.2 F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

** For Unit 2 Cycle 10 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, the calculated value of F^T_{xy} shall be increased by 1% prior to comparison with the limit.

* See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days* of accumulated operation in **MODE** 1, and

c. Within four hours if the **AZIMUTHAL POWER TILT** (T_q) is outside the limit provided in Specification 3.2.4.

d. At least once per 3 days of accumulated operation in **MODE** 1 when monitoring linear heat rate using the Excore Detector Monitoring System per Surveillance 4.2.1.3.

4.2.2.1.3 F^T_{xy} shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all CEAs at or above the Long Term Steady State Insertion Limit. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects.

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* For Unit 2 Cycle 10 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, this surveillance shall be performed at least once per 15 days of accumulated operation in **MODE** 1.

3/4.2.3 TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be within the limits provided in the COLR.**

APPLICABILITY: **MODE** 1*.

ACTION: With F_r^T outside the limits provided in the COLR, within 6 hours either:

a. Be in at least **HOT STANDBY**, or

b. Withdraw and maintain the CEAs at or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6 and reduce **THERMAL POWER** as follows:

   1. Reduce **THERMAL POWER** to bring the combination of **THERMAL POWER** and F_r^T within the limits provided in the COLR, and maintain the peripheral axial shape index within the DNB axial flux offset control limits provided in the COLR, or

   2. Reduce **THERMAL POWER** to less than or equal to the limit established by the full core power distribution monitoring system as a function of F_r^T.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of **RATED THERMAL POWER** after each fuel loading,

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** For Unit 2 Cycle 10 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, the calculated value of F_r^T shall be increased by 1% prior to comparison with the limit.

* See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31* days of accumulated operation in **MODE** 1, and

c. Within four hours if the **AZIMUTHAL POWER TILT** (T_q) is outside the limit given in Specification 3.2.4.

4.2.3.3 F_r^T shall be determined each time a calculation is required by using the incore detectors to obtain a power distribution map with all CEAs at or above the Long Term Steady State Insertion Limit.

* For Unit 2 Cycle 10 only, when the percentage of **OPERABLE** incore detector locations (e.g., strings) falls below 75%, this surveillance shall be performed at least once per 15 days of accumulated operation in **MODE** 1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.4 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The **AZIMUTHAL POWER TILT** (T_q) shall not exceed 0.030.

APPLICABILITY: **MODE** 1 above 50% of **RATED THERMAL POWER***.

ACTION:

a. With the indicated **AZIMUTHAL POWER TILT** determined to be > 0.030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the Total Planar Radial Peaking Factor (F^T_{xy}) and the Total Integrated Radial Peaking Factor (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated **AZIMUTHAL POWER TILT** determined to be > 0.10, operation may proceed for up to 2 hours provided that the Total Integrated Radial Peaking Factor (F^T_r) and Total Planar Radial Peaking Factor (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the **THERMAL POWER** level is restricted to ≤ 20% of **RATED THERMAL POWER**.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The **AZIMUTHAL POWER TILT** shall be determined to be within the limit by:

a. Calculating the tilt at least once per 12 hours, and

b. Using the incore detectors to determine the **AZIMUTHAL POWER TILT** at least once per 12 hours when one excore channel is inoperable and **THERMAL POWER** is > 75% of **RATED THERMAL POWER**.

* See Special Test Exception 3.10.2.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown:

a. Cold Leg Temperature ≤ 548°F

b. Pressurizer Pressure ≥ 2200 psia*

c. Reactor Coolant System Total Flow Rate ≥ 370,000 gpm

d. **AXIAL SHAPE INDEX, THERMAL POWER** as specified in the COLR

APPLICABILITY: **MODE** 1.

ACTION: With any of the above parameters exceeding its limit, restore the parameter 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

4.2.5.1 Each of the parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

* Limit not applicable during either **THERMAL POWER** ramp increase in excess of 5% of **RATED THERMAL POWER** per minute or a **THERMAL POWER** step increase of greater than 10% of **RATED THERMAL POWER**.

LIMITING CONDITION FOR OPERATION (Continued)

c. For monitoring the **TOTAL PLANAR RADIAL PEAKING FACTOR**, the **TOTAL INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate:

1. At least 75%* of all incore detector locations,

2. A minimum of 9 **OPERABLE** incore detector segments at each detector segment level, and

3. A minimum of 2 **OPERABLE** detector segments in the inner 109 fuel assemblies and 2 **OPERABLE** segments in the outer 108 fuel assemblies at each segment level.

An **OPERABLE** incore detector segment shall consist of an **OPERABLE** rhodium detector constituting one of the segments in a fixed detector string.

An **OPERABLE** incore detector location shall consist of a string in which at least three of the four incore detector segments are **OPERABLE**.

An **OPERABLE** quadrant symmetric incore detector segment group shall consist of a minimum of three **OPERABLE** rhodium incore detector segments in 90° symmetric fuel assemblies.

APPLICABILITY: When the Incore Detection System is used for:

a. Monitoring the **AZIMUTHAL POWER TILT**,

b. Recalibration of the Excore Neutron Flux Detection System, or

c. Monitoring the **TOTAL PLANAR RADIAL PEAKING FACTOR**, the **TOTAL INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

ACTION: With the Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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* For Unit 2 Cycle 10 only, the following requirement shall be substituted for Limiting Condition for Operation 3.3.3.2.c.1:

At least 60% of all incore detector locations,

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:

1. Monitoring the **AZIMUTHAL POWER TILT**.

2. Recalibration of the Excore Neutron Flux Detection System.

3. Monitoring the **TOTAL PLANAR RADIAL PEAKING FACTOR**, the **TOTAL INTEGRATED RADIAL PEAKING FACTOR**, or the linear heat rate.

b. At least once per **REFUELING INTERVAL** 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.

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**.*

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 Power Level-High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

b. With one reactor coolant loop and associated steam generator in operation and with one or more main steam line code safety valves associated with the operating steam generator inoperable, operation in **MODE** 3 may proceed provided:

   1. That at least 2 main steam line code safety valves on the non-operating steam generator are **OPERABLE**, and

   2. That within 4 hours the inoperable valve is restored to **OPERABLE** status; otherwise, be in at least **HOT STANDBY** within the next 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

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* Entry into **MODE** 3 is permitted to determine **OPERABILITY** of main steam line code safety valves. During this time, at least two main steam line code safety valves per steam generator shall be **OPERABLE**.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT (PERCENT OF RATED THERMAL POWER)
1	93
2	79
3	66

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER		LIFT SETTINGS* ALLOWABLE	ORIFICE SIZE
a.	RV-3992/4000	935-995 psig	R
b.	RV-3993/4001	935-995 psig	R
c.	RV-3994/4002	935-1035 psig	R
d.	RV-3995/4003	935-1035 psig	R
e.	RV-3996/4004	935-1065 psig	R
f.	RV-3997/4005	935-1065 psig	R
g.	RV-3998/4006	935-1065 psig	R
h.	RV-3999/4007	935-1065 psig	R

* Lift settings for a given steam line are also acceptable if any 2 valves lift between 935 and 995 psig, any 2 other valves lift between 935 and 1035 psig, and the 4 remaining valves lift between 935 and 1065 psig.

3/4.7 **PLANT SYSTEMS**

3/4.7.1 TURBINE CYCLE

Auxiliary Feedwater System

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two auxiliary feedwater trains consisting of one steam-driven and one motor-driven pump and associated flow paths capable of automatically initiating flow shall be **OPERABLE**. (An **OPERABLE** steam-driven train shall consist of one pump aligned for automatic flow initiation and one pump aligned in standby.)*

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

ACTION:

a. With any single pump inoperable, perform the following:

1. With No. 23 motor-driven pump inoperable:

   (a) Align the standby steam-driven pump to automatic initiating status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours, and

   (b) Restore No. 23 motor-driven pump to **OPERABLE** status within the next 7 days or be in **HOT SHUTDOWN** within the next 12 hours.

2. With one steam-driven pump inoperable:

   (a) Align the **OPERABLE** steam-driven pump to automatic initiating status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours, and

   (b) Restore the inoperable steam-driven pump to standby status (or automatic initiating status if the other steam-driven pump is to be placed in standby) within the next 7 days or be in **HOT SHUTDOWN** within the next 12 hours.

b. With any two pumps inoperable:

1. Verify that the remaining pump is aligned to automatic initiating status within one hour, and

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* A standby pump shall be available for operation but aligned so that automatic flow initiation is defeated upon AFAS actuation.

LIMITING CONDITION FOR OPERATION (Continued)

2. Verify within one hour that No. 13 motor-driven pump is **OPERABLE** and valve 1-CV-4550 has been exercised within the last 30 days, and

3. Restore a second pump to automatic initiating status within 72 hours or be in **HOT SHUTDOWN** within the next 12 hours.

c. Whenever a subsystem(s) (a subsystem consisting of one pump, piping, valves and controls in the direct flow path) required for operability is inoperable for the performance of periodic testing (e.g., manual discharge valve closed for pump Total Dynamic Head Test or Logic Testing) a dedicated operator(s) will be stationed at the local station(s) with direct communication to the Control Room. Upon completion of any testing, the subsystem(s) required for operability will be returned to its proper status and verified in its proper status by an independent operator check.

d. The requirements of Specification 3.0.4 are not applicable whenever one motor and one steam-driven pump (or two steam-driven pumps) are aligned for automatic flow initiation.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater flowpath shall be demonstrated **OPERABLE**:

a. At least once per 31 days by:

1. Verifying that each steam-driven pump develops a Total Dynamic Head of $\geq$ 2800 ft. on recirculation flow. (If verification must be demonstrated during **STARTUP**, surveillance testing shall be performed upon achieving an RCS temperature $\geq$ 300°F and prior to entering **MODE** 1).

2. Verifying that the motor-driven pump develops a Total Dynamic Head of $\geq$ 3100 ft. on recirculation flow.

3. Cycling each testable, remote-operated valve that is not in its operating position through at least one complete cycle.

4. Verifying that each valve (manual, power-operated or automatic) in the direct flow path that is not locked, sealed or otherwise secured in position is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

b. Before entering **MODE** 3 after a **COLD SHUTDOWN** of at least 14 days by completing a flow test that verifies the flow path from the condensate storage tank to the steam generators.

c. At least once per 18 months by:

1. Verifying that each automatic valve in the flow path actuates to its correct position (verification of flow-modulating characteristics not required) and each auxiliary feedwater pump automatically starts upon receipt of each AFAS test signal, and

2. Verifying that the Auxiliary Feedwater System is capable of providing a minimum of 300 gpm nominal flow to each flow leg*.

* This surveillance may be performed on one flow leg at a time.

3/4.7.1 TURBINE CYCLE

Condensate Storage Tank

LIMITING CONDITION FOR OPERATION

3.7.1.3 The No. 12 condensate storage tank (CST) shall be **OPERABLE** with a minimum contained water volume of 150,000 gallons per unit.

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

ACTION: With the No. 12 condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to **OPERABLE** status or be in **HOT SHUTDOWN** within the next 12 hours, or

b. Demonstrate the **OPERABILITY** of the No. 21 condensate storage tank as a backup supply to the auxiliary feedwater pumps and restore the No. 12 condensate storage tank to **OPERABLE** status within 7 days or be in **HOT SHUTDOWN** within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The No. 12 condensate storage tank shall be demonstrated **OPERABLE** at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The No. 21 condensate storage tank shall be demonstrated **OPERABLE** at least once per 12 hours by verifying that the tank contains a minimum of 150,000 gallons of water and by verifying that the flow path for taking suction from this tank is **OPERABLE** with the manual valves in this flow path open whenever the No. 21 condensate storage tank is the supply source for the auxiliary feedwater pumps.

3/4.7 **PLANT SYSTEMS**

3/4.7.1 TURBINE CYCLE

Activity

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be $\leq$ 0.10 μCi/gram **DOSE EQUIVALENT I-131.**

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

ACTION: With the specific activity of the Secondary Coolant System > 0.10 μCi/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 4.7-2.

TABLE 4.7-2

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.

3/4.7.1 TURBINE CYCLE

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 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, **POWER OPERATION** may continue provided the inoperable valve is either restored to **OPERABLE** status or closed within 4 hours; otherwise, be in **HOT SHUTDOWN** within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in **MODES** 1, 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in **HOT SHUTDOWN** within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated **OPERABLE** by verifying full closure in less than 5.2 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 90°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION: With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to $\leq$ 200 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generators shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant < 90°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two component cooling water loops shall be **OPERABLE**. At least one component cooling water heat exchanger shall be operating and the remaining component cooling water heat exchanger may be in standby.

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

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

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops 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 **REFUELING INTERVAL** during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection Actuation test signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent service water loops shall be **OPERABLE**.

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

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

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two service water loops 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 **REFUELING INTERVAL** during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on Safety Injection Actuation and Containment Spray Actuation test signals.

3/4.7.5 SALTWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 At least two independent saltwater loops shall be **OPERABLE**.

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

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 At least two saltwater loops shall be **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 **REFUELING INTERVAL** during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection Actuation test signal.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The Control Room Emergency Ventilation System shall be **OPERABLE** with:

a. Two filter trains,

b. Two air conditioning units,

c. Two isolation valves in each Control Room outside air intake duct,

d. Two isolation valves in the common exhaust to atmosphere duct, and

e. One isolation valve in the toilet area exhaust duct.

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

ACTION:

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

b. With one air conditioning unit inoperable, restore the inoperable unit 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.

c. With one isolation valve per Control Room outside air intake duct inoperable, operation may continue provided the other isolation valve in the same duct is maintained closed; otherwise, be in at least **HOT STANDBY** within 6 hours and in **COLD SHUTDOWN** within the following 30 hours.

d. With one common exhaust to atmosphere duct isolation valve inoperable, restore the inoperable valve 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.

e. With the toilet area exhaust duct isolation valve inoperable, restore the inoperable valve 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

4.7.6.1 The Control Room Emergency Ventilation System shall be demonstrated **OPERABLE**:

a. At least once per 62 days, on a **STAGGERED TEST BASIS**, by deenergizing the backup Control Room air conditioner and verifying that the emergency Control Room air conditioners maintain the air temperature $\leq$ 104°F for at least 12 hours when in the recirculation mode.

b. At least once per 31 days by initiation flow through each HEPA filter and charcoal adsorber train and verifying that each train operates for at least 15 minutes.

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

   1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

   2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow rate of 2000 cfm $\pm$ 10%.

   3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying a system flow rate of 2000 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 from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the filter train shall be demonstrated **OPERABLE** by also verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow of 2000 cfm ± 10%.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

2. Verifying that on a control room high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks and that both of the isolation valves in each inlet duct and common exhaust duct, and the isolation valve in the toilet area exhaust duct, close.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter system at a flow rate of 2000 cfm $\pm$ 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter system at a flowrate of 2000 cfm $\pm$ 10%.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The ECCS Pump Room Exhaust Ventilation System shall be **OPERABLE** with one HEPA filter and charcoal adsorber train and two exhaust fans.

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

ACTION:

a. With one ECCS pump room exhaust fan inoperable, restore the inoperable fan 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.

b. With the ECCS exhaust filter train inoperable, restore the filter train 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

4.7.7.1 The ECCS Pump Room Exhaust Ventilation System shall be demonstrated **OPERABLE**:

a. At least once per 31 days by initiating, from the Control Room, flow through the HEPA filter and charcoal adsorber train and verifying that each exhaust fan operates for at least 15 minutes.

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 charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

4. Verifying a system flow rate of 3000 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 from an adsorber tray or from an adsorber test tray in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodine when the sample is tested in accordance with ANSI N510-1975 (30°C, 95% R.H.).

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the filter train shall be demonstrated **OPERABLE** by also verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the ventilation system at a flow rate of 3000 cfm ± 10%.

d. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the filter train at a flow rate of 3000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they tested in-place in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, while operating the filter train at a flow rate of 3000 cfm $\pm$ 10%.

g. After maintenance affecting the air flow distribution by testing in-place and verifying that the air flow distribution is uniform within $\pm$ 20% of the average flow per unit when tested in accordance with the provisions of Section 9 o; "Industrial Ventilation" and Section 8 of ANSI N510-1975.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All safety-related snubbers[1] shall be **OPERABLE**.

APPLICABILITY: **MODES** 1, 2, 3 and 4. (**MODES** 5 and 6 for snubbers located on systems required **OPERABLE** in those **MODES**.)

ACTION: With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to **OPERABLE** status, and perform an engineering evaluation* per Specification 4.7.8.1.b and c on the supported component or declare the supported system inoperable and follow the appropriate **ACTION** statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each snubber shall be demonstrated **OPERABLE** by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5. As used in this Specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

1 Safety-related snubbers include those snubbers installed on safety-related systems and snubbers on non-safety-related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety-related system.

* A documented, visual inspection shall be sufficient to meet the requirements for an engineering evaluation. Additional analyses, as needed, shall be completed in a reasonable period of time.

a. Visual Inspections

Visual inspections shall be performed in accordance with the schedule determined by Table 4.7-3. Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently or jointly according to the schedule determined by Table 4.7-3. The visual inspection interval for each population or category of snubbers shall be determined based upon the criteria provided in Table 4.7-3 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment 139.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired **OPERABILITY**, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. 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, remedied and functionally tested for that particular snubber and for other snubbers that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined **OPERABLE** per Specification 4.7.8.1.d, as applicable. When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined **OPERABLE** via functional testing for the purpose of establishing the next visual inspection interval.

SURVEILLANCE REQUIREMENTS (Continued)

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

c. Functional Tests

At least once per **REFUELING INTERVAL**, a representative sample of 10% of each type of snubbers in use in the plant shall be functionally tested either in-place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8.1.d, an additional 5% of that type snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

Snubbers identified as "Especially Difficult to Remove" or in "High Radiation Zones" shall also be included in the representative sample.**

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period. Failure of these snubbers shall not entail functional testing of additional snubbers.

** Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

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 generically susceptible snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) which are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

SURVEILLANCE REQUIREMENTS (Continued)

At least once per **REFUELING INTERVAL****, the installation and maintenance records for each safety-related snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review*. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

** The first snubber service life review following Amendment No. 145 shall be performed within 18 months* of the previous review.

* The provisions of Specification 4.0.2 are applicable.

TABLE 4.7-3

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF INOPERABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

TABLE 4.7-3 (Continued)

TABLE NOTATION

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of inoperable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of inoperable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of inoperable snubbers as determined by interpolation.

Note 3: If the number of inoperable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of inoperable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of inoperable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of inoperable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of inoperable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 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 $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

1. Either decontaminated and repaired, or

2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.9.1.2 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.

SURVEILLANCE REQUIREMENTS (Continued)

a. Sources in use - At least once per six 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.

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

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

4.7.9.1.3 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 $\geq$ 0.005 microcuries of removable contamination.

3/4.7.10 WATERTIGHT DOORS

LIMITING CONDITION FOR OPERATION

3.7.10 The following watertight doors shall be closed except when the door is being used for normal entry and exit:

a. ECCS Pump Room Doors (4).

b. Service Water Pump Room to Heater Bay Doors (2).

c. Auxiliary Feed Pump Room to Heater Bay Doors (2).

d. Emergency Escape Hatch, Service Water Pump Room from Penetration Room.

e. Main Steam Piping Area from Piping Penetration Room Door.

f. Passage to Main Steam Piping Area Door.

g. Warehouse to Intake Structure Door, Elevation 12'.

h. Outside to Intake Structure Door.

i. Warehouse to Intake Structure Door Elevation 29'.

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

ACTION: With one or more of the above doors open, restore the door to its closed position 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

4.7.10 The above watertight doors shall be determined closed at least once per 12 hours.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Fire Suppression Water System

LIMITING CONDITION FOR OPERATION

3.7.11.1 The Fire Suppression Water System shall be **OPERABLE** with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Two water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An **OPERABLE** flow path capable of taking suction from the Pretreated Water Storage Tanks Numbers 11 and 12 and transferring the water through distribution piping with **OPERABLE** sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser required to be **OPERABLE** per Specifications 3.7.11.2, 3.7.11.4, and 3.7.11.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to **OPERABLE** status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable:

1. Establish a backup Fire Suppression Water System within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.9.2:

   a) By telephone within 24 hours,

LIMITING CONDITION FOR OPERATION (Continued)

b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

c) In writing 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.

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 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 on a **STAGGERED TEST BASIS** by starting the electric motor-driven pump and operating it for at least 15 minutes. This test shall be performed on a **STAGGERED TEST BASIS** with the test required by 4.7.11.1.2.a.2.

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.

d. At least once per 12 months by performance of a system flush of the filled portions of the system.

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 automatic valve in the flow path actuates to its correct position,

2. Verifying that each pump develops at least 2500 gpm at a discharge pressure of 125 psig,

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each high pressure pump starts (sequentially) to maintain the Fire Suppression Water System pressure $\geq$ 80 psig.

g. At least once per **REFUELING INTERVAL** by: (1) 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, and (2) performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence and cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

4.7.11.1.2 The 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 174 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes. This test shall be performed on a **STAGGERED TEST BASIS** with the test required by Specification 4.7.11.1.1.b.

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-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for viscosity, water and sediment.

c. At least once per 18 months by:

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

2. Verifying the diesel starts from ambient conditions on the auto-start signal and operates for $\geq$ 20 minutes while loaded with the fire pump.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.1.3 The fire pump diesel starting 24-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 $\geq$ 24 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 anti-corrosion material.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Spray and/or Sprinkler Systems

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems shown in Table 3.7-5 shall be **OPERABLE**:

APPLICABILITY: Whenever equipment in the spray/sprinkler protected areas is required to be **OPERABLE**.

ACTION:

a. With one or more of the required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant safe shutdown systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 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, not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 12 months by cycling each 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 verifying that the automatic valves in the flow path actuate to their correct positions on a simulated test signal.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the area in the vicinity of each nozzle(s) to verify the spray pattern will not be obstructed.

TABLE 3.7-5

FIRE PROTECTION SPRINKLERS
UNIT 2

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
Unit 2 Aux Feed Pump Room 605*	12'-0"
Unit 2 East Piping Area Room 408*	45'-0"
Unit 2 East Elec Pen Room 409*	45'-0"
Unit 2 West Elec Pen Room 414*	45'-0"
Cable Chase 2A*	45'-0"
Cable Chase 2B*	45'-0"
Unit 2 Main Steam Piping Room 309*	45'-0"
Unit 2 Component Cooling Pp Room 201	5'-0"
Unit 2 East Piping Area 203*	5'-0"
Unit 2 Rad Exh Vent Equip Room 204*	5'-0"
Unit 2 Service Water Pp Room 205*	5'-0"
Unit 2 Boric Acid Tk and Pp Room 215*	5'-0"
Unit 2 Reactor Coolant Makeup Pump Room 216A*	5'-0"
Unit 2 Charging Pump Room 105*	(-)10'-0"
Unit 2 Misc Waste Monitor Tk Room 106*	(-)10'-0"
Unit 2 ECCS Pump Room 101*	(-)15'-0"
21 Diesel Generator	45'-0"
Unit 2 East Pipe Pen Room 206/310*	5'-0"

* Sprinklers required to ensure the **OPERABILITY** of redundant safe shutdown equipment.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Halon Systems

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following Halon Systems shall be **OPERABLE** with the storage tanks having at least 95% of full charge weight (or level) and 90% of full charge pressure.

a. Cable spreading room total flood system, and associated vertical cable chase 1C, Unit 2.

b. 4160 volt switchgear room 27' & 45' elevation Unit 2.

APPLICABILITY: Whenever equipment protected by the Halon System is required to be **OPERABLE**.

ACTION:

a. With both the primary and backup Halon Systems protecting the areas inoperable, within one hour establish an hourly fire watch with backup fire suppression equipment for those areas protected by the inoperable Halon System. Restore the system to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3 Each of the above required Halon 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 is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 12 months by performing a visual inspection of the nozzle(s) and visible flow paths for obstructions.

d. At least once per 18 months by verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

e. Following completion of major maintenance or modifications on the system(s), within 72 hours by performance of a flow test through headers and nozzles to assure no blockage.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Fire Hose Stations

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-6 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 3.7-6 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an **OPERABLE** hose station within 1 hour. Restore the fire hose station(s) to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the fire hose station(s) to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated **OPERABLE**:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station. Hose stations located in the containment shall be visually inspected on each scheduled reactor shutdown, but not more frequently than every 31 days.

b. At least once per 18 months for hose stations located outside the containment and once per **REFUELING INTERVAL** for hose stations inside the containment by:

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

   2. Replacement of all degraded gaskets in couplings.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years for hose stations located outside the containment and once per **REFUELING INTERVAL** for hose stations inside the containment 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 at least 50 psig greater than the maximum pressure available at that hose station or replacement with a new hose.

TABLE 3.7-6

FIRE HOSE STATIONS
UNIT 2

LOCATION	**ELEVATION**	**NUMBER OF HOSE STATIONS**
1. Containment	10'	2
	45'	2
	69'	2
2. Auxiliary Building	-15'*	1**
	-10'*	2**
	5'	3
	27'	2
	45'	4
	69'*	3
3. Turbine Building, Heater Bay Outside Service Water Pump Rooms and Aux Feedwater Pump Rooms	12'	2
Outside Switchgear Room	27'	1
Outside Switchgear Room	45'	2
4. Intake Structure	10'*	1

* Fire Hose Stations required for primary protection to ensure the **OPERABILITY** of safety related equipment.

** Hose Stations which serve both Units 1 and 2.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

Yard Fire Hydrants and Hydrant Hose Houses

LIMITING CONDITION FOR OPERATION

3.7.11.5 The following yard fire hydrants and associated hydrant hose houses shall be **OPERABLE**:

a. #6 yard hydrant and associated hydrant hose house, which provides primary protection for Unit 2 RWT blockhouse.

b. #7 yard hydrant and associated hydrant hose house, which provides primary protection for Unit 1 RWT blockhouse.

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 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 hydrant or associated hydrant hose house is the primary means of fire suppression. Restore the hydrant or hose house to **OPERABLE** status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the hydrant or hose house to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.5 Each of the yard fire hydrants and associated hydrant hose houses 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 6 months (once during March, April or May and once during September, October or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any yard fire hydrant.

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**.

3/4.7 **PLANT SYSTEMS**

3/4.7.12 PENETRATION FIRE BARRIERS

LIMITING CONDITION FOR OPERATION

3.7.12 All fire barrier penetrations (i.e., cable penetration barriers, fire doors and fire dampers), in fire zone boundaries, protecting safe shutdown areas shall be **OPERABLE**.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations inoperable within one hour either establish a continuous fire watch on at least one side of the affected penetration, or verify the **OPERABILITY** of fire detectors on at least one side of the inoperable fire barrier and establish an hourly fire watch patrol; or verify the operability of Automatic Sprinkler Systems (including the water flow alarm and supervisory system) on both sides of the inoperable fire barrier. Restore the inoperable fire barrier penetration(s) to **OPERABLE** status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperable penetration and plans and schedule for restoring the fire barrier penetration(s) to **OPERABLE** status.

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

SURVEILLANCE REQUIREMENTS

4.7.12 Each of the above required fire barrier penetrations shall be verified to be **OPERABLE**:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a fire barrier penetration to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration(s).

3/4.9 **REFUELING OPERATIONS**

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling pool shall be maintained within the limit provided in the COLR.

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 $\geq$ 40 gpm of 2300 ppm boric acid solution or its equivalent until the boron concentration is within its limit. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration shall be determined to be within its limit 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.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling pool shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

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.

The most limiting **SHUTDOWN MARGIN** requirement at beginning of cycle is determined by the requirements of several transients, including Boron Dilution and Steam Line Rupture. The **SHUTDOWN MARGIN** requirements for these transients are relatively small and nearly the same. However, the most limiting **SHUTDOWN MARGIN** requirement at end of cycle comes from just one transient, the Steam Line Rupture event. The requirement for this transient at end of cycle is significantly larger than that for any other event at that time in cycle and, also, considerably larger than the most limiting requirement at beginning of cycle.

Adherence to Technical Specification 3.1.1.1 provides assurance that the available **SHUTDOWN MARGIN** at any time in cycle will exceed the most limiting **SHUTDOWN MARGIN** requirement at that time in cycle. Without the specified **SHUTDOWN MARGIN** available, immediate boration is required (by Specifications 3/4.1.1.1 or 3/4.1.1.2) that is at least equivalent to boration from the refueling water tank, at its minimum boric acid concentration, via a charging pump, at its minimum flow rate. For example, lower flow rates with higher boric acid concentrations could also provide the equivalent boration, but should be verified as equivalent prior to use. When in **MODE** 1 or **MODE** 2 with $K_{eff} \geq 1.0$, operation within the CEA Group Insertion Limits assures that there is sufficient available **SHUTDOWN MARGIN** to match the **SHUTDOWN MARGIN** requirements in the safety analysis.

In **MODE** 5, the reactivity transients resulting from any event are minimal and do not vary significantly during the cycle. The most limiting event at any time during the cycle is a Boron Dilution Event with the pressurizer level less than 90 inches and the sources of non-borated water restricted. Adherence to Technical Specification 3.1.1.2 provides assurance that the available **SHUTDOWN MARGIN** will exceed the most limiting **SHUTDOWN MARGIN** requirement at any time in cycle.

The components required to perform this function include: 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from **OPERABLE** diesel generators. At or below 80% of **RATED THERMAL POWER**, there is a corresponding decrease in decay heat which compensates for the loss of injection from one charging pump assumed in the Small Break LOCA Analyses.

With the RCS average temperature above 200°F, a minimum of two independent and redundant boration 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 will provide sufficient **SHUTDOWN MARGIN** from all operating conditions assuming xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid tanks, the concentration and volume of which are met by the range of values given in Specifications 3.1.2.8 and 3.1.2.9, or 55,627 gallons of 2300 ppm borated water from the refueling water tank. However, to be consistent with the ECCS requirements, the RWT is required to have a minimum contained volume of 400,000 gallons during **MODES** 1, 2, 3 and 4. The maximum boron concentration of the refueling water tank shall be limited to 2700 ppm and the maximum boron concentration of the boric acid storage tanks shall be limited to 8% to preclude the possibility of boron precipitation in the core during long term ECCS cooling.

With the RCS temperature below 200°F, one boration 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 change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing sufficient **SHUTDOWN MARGIN** after xenon decay and cooldown from 200°F to 140°F. This condition requires either boric acid solution from the boric acid tanks, the requirements of which are met by Specification 3.1.2.7, or 9,844 gallons of 2300 ppm borated water from the refueling water tank.

The **OPERABILITY** of one boration system during **REFUELING** ensures that this system is available for reactivity control while in **MODE** 6.

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 maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

For a single CEA misalignment, the time allowance to realign the CEA is permitted for the following reasons:

1. The margin calculations which support the power distribution LCOs for DNBR are based on a steady-state F_r^T as specified in the Technical Specification 3.2.3.

2. When the actual F_r^T is less than the Technical Specification value, additional margin exists.

3. This additional margin can be credited to offset the increase in F_r with time that will occur following a CEA misalignment due to xenon redistribution.

4. If an F_r^T measurement has not been taken recently (within 5 days), a pre-misaligned value of 1.70 is assumed and no time for realignment is permitted.

The requirement to reduce power level after the time limit is reached offsets the continuing increase in F_r^T that can occur due to xenon redistribution. A power reduction is not required below 50% power. Below 50% power there is sufficient conservatism in the DNB power distribution LCOs to completely offset any, or any additional, xenon redistribution effects.

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 brings 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.

There are five different operating modes for control of CEAs; Off, Manual Individual, Manual Group, Manual Sequential and Automatic. The Manual Sequential mode is applicable to only the regulating CEAs and the Automatic mode is disabled and not used for both regulating and shutdown CEAs.

OPERABILITY of the CEA position indicators is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. 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 **OPERABILITY**

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 Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits.

Normally, the Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits. The setpoints for these alarms include allowances described in the COLR.

The control channel excore detectors, in conjunction with the Power Ratio Recorder Monitoring System, can also perform this function by continuously monitoring the **AXIAL SHAPE INDEX** and verifying that the **AXIAL SHAPE INDEX** is maintained within the allowable limits. In conjunction with the use of the Excore Monitoring System and in establishing the **AXIAL SHAPE INDEX** limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the **AZIMUTHAL POWER TILT** restrictions of Specification 3.2.4 are satisfied, and 3) the **TOTAL PLANAR RADIAL PEAKING FACTOR** does not exceed the limits of Specification 3.2.2.

3/4.2.2, 3/4.2.3, and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} AND F^T_r AND **AZIMUTHAL POWER TILT** - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, F^T_r, or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the **ACTION** statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local

BASES

Power Density - High LCOs and LSSS setpoints remain valid. An **AZIMUTHAL POWER TILT** > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The surveillance requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T, F_r^T after each fuel loading prior to exceeding 70% of **RATED THERMAL POWER** provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNB Specified Acceptable Fuel Design Limit (SAFDL) throughout each analyzed transient.

In addition to the DNB criterion, there are two other criteria which influence the DNB axial flux offset control limits. The second criterion is to ensure that the existing core power distribution at full power is less severe than the power distribution factored into the small-break LOCA analysis. This results in a limitation on the allowed negative **AXIAL SHAPE INDEX** value at full power. The third criterion is to maintain limitations on peak linear heat rate at low power levels resulting from Anticipated Operational Occurrences (AOOs). The DNB axial flux offset control limits are used to assure the LHR criterion for this condition because the linear heat rate LCO, for both excore and incore monitoring, is set to maintain only the LOCA kw/ft requirements which are limiting at high power levels. At reduced power levels, the kw/ft requirements of certain AOOs (e.g., CEA withdrawal), tend to become more limiting than that for LOCA.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.

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% of its design pressure of 1000 psia during the most severe anticipated system operational transient. The total relieving capacity for all valves on all of the steam lines is 12.18 x 10^6 lbs/hr at 100% **RATED THERMAL POWER**. 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 main steam line code safety valves are tested and maintained in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code.

In **MODE** 3, two main steam safety valves are required **OPERABLE** per steam generator. These valves will provide adequate relieving capacity for removal of both decay heat and reactor coolant pump heat from the Reactor Coolant System via either of the two steam generators. This requirement is provided to facilitate the post-overhaul setting and **OPERABILITY** testing of the safety valves which can only be conducted when the RCS is at or above 500°F. It allows entry into **MODE** 3 with a minimum number of main steam safety valves **OPERABLE** so that the set pressure for the remaining valves can be adjusted in the plant. This is the most accurate means for adjusting safety valve set pressures since the valves will be in thermal equilibrium with the operating environment.

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:

$$SP = \frac{(X) - (Y)(V)}{X} \times 106.5$$

where:

SP = reduced reactor trip setpoint in percent of **RATED THERMAL POWER**

V = maximum number of inoperable safety valves per steam line

106.5 = Power Level - High Trip Setpoint

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 Auxiliary Feedwater System

The **OPERABILITY** of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of offsite power. A delivered flow of 300 gpm is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 300°F when the Shutdown Cooling System may be placed into operation.

Flow control valves were installed in the system in order to allow automatic flow initiation to a value selected by the Operator. Maximum flow to the steam generators from the motor driven AFW pump powered from the diesel is 300 gpm when feeding both generators (i.e., 150 gpm per leg maximum flow). The flow control valves installed in each leg supplied from the motor driven AFW pump shall be set at a flow setpoint not to exceed 150 gpm per leg. If the flow is only being directed to one steam generator, it is acceptable to deliver a maximum of 330 gpm because the flow error associated with the non-used loop is eliminated. These motor driven AFW pump capacity limits are imposed to prevent exceeding the emergency diesel generator load limit. If diesel generator loading is not a limiting concern, the delivered flow from the motor driven AFW pump may be increased up to a maximum of 575 gpm (motor HP limit vice diesel loading limit). These upper flow limits do not apply to the steam driven pumps.

Auxiliary Feedwater flow and response times are conservatively accounted for in the analyses of Design Basis Events. In Main Steam Line Break and Excess Load analyses where Auxiliary Feedwater flow would increase the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is minimized and Auxiliary Feedwater flow is maximized. In Feedline Break, Loss of Feedwater, and Loss of Non-Emergency AC Power analyses, in which Auxiliary Feedwater flow would decrease the consequences of the accidents, the delay time for Auxiliary Feedwater actuation is maximized and Auxiliary Feedwater flow is minimized.

At 10 minutes after an Auxiliary Feedwater Actuation Signal, the operator is assumed to be available to increase or decrease auxiliary feedwater flow to that required by the existing plant conditions.

3/4.7.1.3 Condensate Storage Tank

The **OPERABILITY** of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at **HOT STANDBY** conditions for 6 hours with steam discharge to atmosphere with concurrent and total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 Activity

The limitations on Secondary System specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 Main Steam Line Isolation Valves

The **OPERABILITY** of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The **OPERABILITY** of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses. The main steam isolation valves are surveilled to close in less than 5.2 seconds to ensure that under reverse steam flow conditions, the valves will close in less than the 6.0 seconds assumed in the accident analysis.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 90°F and 200 psig are based on steam generator secondary side limitations and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

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

3/4.7.4 SERVICE WATER SYSTEM

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

3/4.7.5 SALTWATER SYSTEM

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

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The **OPERABILITY** of the Control Room Emergency Ventilation 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 10 CFR Part 50, Appendix A, General Design Criteria 19.

3/4.7.7 ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM

The **OPERABILITY** of the ECCS Pump Room Exhaust Air Filtration System ensures that radioactive materials leaking from 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 effects on offsite dosage calculations was assumed in the accident analyses.

3/4.7.8 SNUBBERS

All safety related 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 non-safety 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.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during the previous inspection, the total population or category size, and the previous inspection interval.

Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and that decision shall be used as the basis upon which to determine the next inspection interval for that category. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

A snubber is considered inoperable if it fails to satisfy the acceptance criteria of the visual inspection. When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system. Operation may continue indefinitely if an engineering review and evaluation can document within 72 hours that the equipment connected to the snubber can continue to perform its required function(s) with the snubber inoperable. If the review and evaluation cannot justify that the supported equipment will perform its required function(s), the equipment must be declared inoperable and the applicable action requirements met.

The Specification allows inspection intervals to be compatible with a 24 month fuel cycle, up to and including an increase to every other refueling outage. To provide assurance of snubber functional reliability, a representative sample of the installed snubbers of each type* will be functionally tested during plant shutdowns at **REFUELING INTERVALS**. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated 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. The service life program is designed to uniquely reflect the conditions at Calvert Cliffs. The criteria for evaluating service life shall be determined, and documented, by the licensee. Records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

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.

3/4.7.10 WATERTIGHT DOORS

This specification is provided to ensure the protection of safety related equipment from the effects of water or steam escaping from ruptured pipes or components in adjoining rooms.

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* Small bore ($\leq$ 8") and large bore (> 8") hydraulic snubbers are examples of different types of snubbers.

3/4.7.11 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 and fire hose stations. 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. Where a continuous fire watch is required in lieu of fire protection equipment and habitability due to heat or radiation is a concern, the fire watch should be stationed in a habitable area as close as possible to the inoperable equipment.

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. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.12 PENETRATION FIRE BARRIERS

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, a continuous fire watch is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration 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 volumes having direct access to the reactor vessel. A K_{eff} of no greater than 0.95 which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

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 accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

During **CORE ALTERATIONS** or movement of irradiated fuel within containment, a release of fission product radioactivity to the environment must be prevented. During **MODES** 1, 2, 3, and 4, this is accomplished by maintaining **CONTAINMENT INTEGRITY** as described in LCO 3.6.1. In other situations, the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment from the outside atmosphere.

The containment structure serves to contain fission product radioactivity which may be released from the reactor core following a Design Basis Accident (DBA), such that offsite radiation exposures are maintained within the requirements of 10 CFR Part 100. Additionally, this structure provides radiation shielding from the fission products which may be present in the containment atmosphere following accident conditions.

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**.

The 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 identified by the annual land use census pursuant to Specification 3.12.2.

6.9.1.9 CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   2.2.1
   3.1.1.1
   3.1.1.2
   3.1.1.4
   3.1.3.1
   3.1.3.6
   3.2.1
   3.2.2.1
   3.2.3
   3.2.5
   3.9.1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC; specifically, those described in the following documents:

   (1) CENPD-199-P, Latest Approved Revision, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986

   (2) CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II," December 1979

   (3) CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," January 1980

   (4) CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2," March 1980

   (5) CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981

(6) Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

(7) CEN-348(B)-P, "Extended Statistical Combination of Uncertainties," January 1987

(8) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

(9) CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986

(10) CENPD-162-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"

(11) CENPD-207-P-A, Latest Approved Revision, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"

(12) CENPD-206-P-A, Latest Approved Revision, "TORC Code, Verification and Simplified Modeling Methods"

(13) CENPD-225-P-A, Latest Approved Revision, "Fuel and Poison Rod Bowing"

(14) CENPD-266-P-A, Latest Approved Revision, "The ROCS and DIT Computer Code for Nuclear Design"

(15) CENPD-275-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"

(16) CENPD-382-P-A, Latest Approved Revision, "C-E Methodology for Core Designs Containing Erbium Burnable Absorbers"

(17) CENPD-139-P-A, Latest Approved Revision, "C-E Fuel Evaluation Model Topical Report"

(18) CEN-161-(B)-P-A, Latest Approved Revision, "improvements to Fuel Evaluation Model"

(19) CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model," April 1989

(20) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

(21) CEN-372-P-A, Latest Approved Revision, "Fuel Rod Maximum Allowable Gas Pressure"

(22) Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"

(23) CENPD-132, Supplement 3-P-A, Latest Approved Revision, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"

(24) CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985

(25) CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985

(26) Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"

(27) CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977

(28) Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"

(29) CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977

(30) CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977

(31) Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"

(32) CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977

(33) Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"

(34) Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"

(35) Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"

(36) Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"

(37) Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"

(38) Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"

(39) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

(40) CENPD-188-A, Latest Approved Revision, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"

(41) The Full Core Power Distribution Monitoring System referenced in Specifications 3.1.3.1, 3.2.2.1, 3.2.3, and the BASES is described in the following documents:

(a) CENPD-153-P, Latest Approved Revision, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"

(b) CEN-199(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2," November 1979

(c) Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"

(d) Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Instrumentation, Specification 3.3.3.4.

d. Seismic Event Analysis, Specification 4.3.3.3.2.

e. Core Barrel Movement, Specification 3.4.11.

f. Fire Detection Instrumentation, Specification 3.3.3.7.

g. Fire Suppression Systems, Specifications 3.7.11.1, 3.7.11.2, 3.7.11.3, 3.7.11.4, and 3.7.11.5.

h. Penetration Fire Barriers, Specification 3.7.12.

i. Steam Generator Tube Inspection Results, Specification 4.4.5.5.a and c.

j. Specific Activity of Primary Coolant, Specification 3.4.8.

k. Containment Structural Integrity, Specification 4.6.1.6.

l. Radioactive Effluents - Calculated Dose and Total Dose, Specifications 3.11.1.2, 3.11.2.2, 3.11.2.3, and 3.11.4.

m. Radioactive Effluents - Liquid Radwaste, Gaseous Radwaste and Ventilation Exhaust Treatment Systems Discharges, Specifications 3.11.1.3 and 3.11.2.4.

n. Radiological Environmental Monitoring Program Specification 3.12.1.

o. Radiation Monitoring Instrumentation, Specification 3.3.3.1 (Table 3.3-6).

p. Overpressure Protection Systems, Specification 3.4.9.3.

q. Hydrogen Analyzers, Specification 3.6.5.1.

r. Post-Accident Instrumentation, Specification 3.3.3.6.

6.10 RECORD RETENTION

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

a. Records and logs of facility 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 reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

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

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

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

a. Records and drawing changes reflecting facility 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 facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components identified in Table 5.7.1.

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

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

i. Records of Quality Assurance activities identified in the NRC approved QA Manual as lifetime records.

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 POSRC, the Procedure Review Committee, and the OSSRC.

l. Records of the service lives of all safety related snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by 10 CFR Part 20.203(c)(2):

a. A high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Special or Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A high radiation area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a, above, and in addition locked barricades shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained by the Supervisor - Radiation Control Operations and the Operations Shift Supervisor on duty under their separate administrative control.

6.13 SYSTEM INTEGRITY

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

a. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

b. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.14 IODINE MONITORING

The licensee shall implement 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:

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* It is acceptable if the licensee maintains details of the program in plant operation manuals (e.g., chemistry procedures, training instructions, maintenance procedures, ERPIPs).

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

6.15 POSTACCIDENT SAMPLING

The licensee shall establish, implement and maintain 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:

a. Training of personnel,

b. Procedures for sampling and analysis, and

c. Provisions for maintenance of sampling and analysis equipment.

6.16 PROCESS CONTROL PROGRAM (PCP)

6.16.1 The **PCP** shall be approved by the Commission prior to implementation.

6.16.2 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. An evaluation supporting the premise that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   2. A reference to the date and the POSRC meeting number in which the changes(s) was reviewed and found acceptable to the POSRC.

b. Shall become effective upon review by the POSRC and approval of the Plant General Manager.

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* It is acceptable if the licensee maintains details of the program in plant operation manuals (e.g., chemistry procedures, training instructions, maintenance procedures, ERPIPs).

6.17 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.17.1 The **ODCM** shall be approved by the Commission prior to implementation.

6.17.2 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. Sufficient information to support the rationale for the change. Information submitted should consist of a package of those pages of the **ODCM** to be changed with each page numbered and provided with a change number and/or change date 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. Documentation of the fact that the change has been reviewed and found acceptable by the POSRC.

b. Shall become effective upon review by the POSRC and approval of the Plant General Manager.

6.18 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.18.1 Licensee initiated major changes to the Radioactive Waste Systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the modification to the waste system is completed. The discussion of each change shall contain:

a. A description of the equipment, components and processes involved.

b. Documentation of the fact that the change including the safety analysis was reviewed and found acceptable by the POSRC.