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Revision 2

POOR ORIGINAL

STANDARD TECHNICAL SPECIFICATIONS FOR GENERAL ELECTRIC BOILING WATER REACTORS

Revision of August 1979

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Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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STANDARD

TECHNICAL SPECIFICATIONS

GENERAL ELECTRIC

BOILING WATER REACTORS

(GE-STS)

REVISION 2 of

August 1979

1069 097

FOREWORD

The information contained in the following paragraphs briefly describes the applicability, format and implementation of the General Electric Standard Technical Specification package.

APPLICABILITY

This Standard Technical Specification (STS) has been structured for the broadest possible use on General Electric plants currently being reviewed for an Operating License. Optional specifications are provided for those features and systems which may be included in individual plant designs but are not generic in their scope of application.

This revision of the GE-STS does not typically include requirements which may be added or revised as a result of the NRC staff's further review of the Three Mile Island incident.

FORMAT

The format of the STS addresses the categories required by 10 CFR 50 and consists of six sections covering the areas of: Definitions, Safety Limits and Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features and Administrative Controls. The Limiting Conditions for Operation and Surveillance Requirements, Sections 3 and 4, are presented in a combined format with each LCO appearing at the top of the page followed immediately by the applicable Surveillance Requirements. The combined Section 3/4 is further subdivided into ten subsections covering the areas of:

1. Reactivity Control Systems
2. Power Distribution Limits
3. Instrumentation
4. Reactor Coolant System
5. Emergency Core Cooling Systems
6. Containment Systems
7. Plant Systems
8. Electrical Power Systems
9. Refueling Operations
10. Special Test Exceptions

The values of those parameters and variables which may vary because of plant design appear as either blanks or parenthesized numbers throughout the STS. The actual value for each parameter will be provided by individual applicants as appropriate for their plants. The values in parentheses are for illustration only.

IMPLEMENTATION

The implementation of the STS on an individual license application will proceed in three phases. The major steps within each phase are indicated below.

Phase I

The applicant should;

1. Obtain copies of the STS from the LPM.
2. Identify and mark those specifications not required because of plant design or other factors. Specifications within this category should be retained in position within the document package for later review and discussion.
3. Identify those areas where specifications are required but are not provided in the STS.
4. Provide the applicable values of the parameters and variables identified by blanks or parentheses in the STS.
5. Provide the figures, graphs and other information required to complete the STS package.

Phase II

1. The Commission staff will review the information provided in the marked up STS package resulting form the Phase I preparation.
2. An applicant/staff meeting will be held to resolve noted differences of position and other related comments from the applicant, vendor and A.E.

Phase III

1. The Commission will provide a Proof and Review edition of the technical specification for final review by all parties based upon the resolution of comments and positions in Phase II.
2. Final comments and corrections will be incorporated into the document as received.
3. The Technical Specifications will be issued by the Commission as Appendix "A" of the Operating License.

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ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be those additional requirements specified as corollary statements to each specification and shall be part of the specification.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

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CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CRITICAL POWER RATIO

1.8. The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

1.13 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection sytems, such as pump seal or valve packing leaks, that is 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 the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.14 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable.

LIMITING CONTROL ROD PATTERN

1.15 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.16 LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.17 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor through and including the actuated device.

MAXIMUM TOTAL PEAKING FACTOR

1.18 The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MINIMUM CRITICAL POWER RATIO

1.19 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

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 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 CONDITION - CONDITION

1.21 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

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 14 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 through a non-isolable fault in a reactor coolant sytem component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.24 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification, 3.6.3.1.

b. All equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

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

e. The sealing mechanism associated with each penetrations; e.g., welds, bellows or O-rings, is OPERABLE.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer to the reactor coolant of (2436) MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.26 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-engergization of the scram pilot valve solenoids.

RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.27 The RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to (recirculation pump breaker trip) from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

REPORTABLE OCCURRENCE

1.28 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

ROD DENSITY

1.29 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.30 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All Reactor Building ventilation system automatic isolation dampers are OPERABLE or secured in the isolated position per Specification 3.6.5.2.

b. The Standby Gas Treatment System is OPERABLE pursuant to Specification 3.6.5.3.

c. At least one door in each access to the Reactor Building is closed.

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

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SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TOTAL PEAKING FACTOR

1.34 The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the core average LHGR.

UNIDENTIFIED LEAKAGE

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

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
N.A.	Not applicable.

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TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown	> 212°F
4. COLD SHUTDOWN	Shutdown	≤ 212°F
5. REFUELING*	Shutdown or Refuel**	≤ 212°F

*Reactor vessel head unbolted or removed and fuel in the vessel
**See Special Test Exception 3.10.3

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than (1.07) with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than (1.07) and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed (1325) psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above (1325) psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to (1325) psig within 2 hours.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the low pressure ECCS, to restore the water level, after depressurizing the reactor vessel, if required.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

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TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ (120) divisions of full scale	≤ ( ) divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale	≤ (15)% of RATED THERMAL POWER	≤ ( )% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale	≤ (0.66 W + (54)%), with ≤ ( )% maximum	≤ ( ), with ≤ ( )% maximum
c. Neutron Flux-Upscale	≤ (120)% of RATED THERMAL POWER	≤ ( )% of RATED THERMAL POWER
3. Reactor Vessel Steam Dome Pressure - High	≤ (1045) psig	≤ ( ) psig
4. Reactor Vessel Water Level - Low, Level 3	≥ (12.5) inches above instrument zero	≥ ( ) inches above instrument zero*
5. Main Steam Line Isolation Valve - Closure	≤ (10)% closed	≤ ( )% closed
6. Main Steam Line Radiation - High	≤ (3) x full power background	≤ ( ) x full power background
7. Primary Containment Pressure - High	≤ (2) psig	≤ ( ) psig
8. Scram Discharge Volume Water Level - High	≤ (50) gallons	≤ ( ) gallons
9. Turbine Stop Valve - Closure	≤ (10)% closed	≤ ( )% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ (500) psig	≥ ( ) psig
11. Reactor Mode Switch in Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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NOTE

The summary statements contained in this section provide the bases for the Specification of Section 2.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36

BASES

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than (1.07). MCPR greater than (1.07) represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no (mechanistic) fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[a], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) Boiling Length (L), GEXL correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia
Mass Flow:	0.1×10^6 to 1.25×10^6 lb/hr-ft^2
Inlet Subcooling:	0 to 100 Btu/lb
Local Peaking:	(1.61) at a corner rod to (1.47) at an interior rod

a. "General Electric BWR Thermal Analysis Bases (GETAB) Data; Correlation and Design application," NEDO-10958-A and NEDE-10958-A.

THERMAL POWER, High Pressure and High Flow (Continued)

Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	(64 Rods in an 8 x 8 array)	

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1, the nominal values of the core parameters listed in Bases Table B2.1.2-2, and the relative assembly power distribution shown in Bases Table B2.1.2-3. Bases Table B2.1.2-4 shows the R-factor distributions that are input to the statistical model which is used to establish the Safety Limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The bases for the uncertainties in the core parameters are given in NEDO-20340[b] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[a]. The power distribution is based on a typical (764) assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in (unit name) during any fuel cycle would not be as severe as the distribution used in the analysis. (The pressure Safety Limits are arbitrarily selected to be the lowest transient overpressures allowed by the applicable codes, ASME Boiler and Pressure Vessel Code, Section III, and USAS Piping Code, Section B31.1.)

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a. "General Electric BWR Thermal Analysis Bases (GETAB) Data, Correlation and design application," NEDO-10958-A and NEDE-10958-A.

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Admendment 1, NEDO-20340-1 dated June 1974 and December 1974, respectively.

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Bases Table B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	(1.76)
Feedwater Temperature	(0.76)
Reactor Pressure	(0.5)
Core Inlet Temperature	(0.2)
Core Total Flow	(2.5)
Channel Flow Area	(3.0)
Friction Factor Multiplier	(10.0)
Channel Friction Factor Multiplier	(5.0)
TIP Readings	(6.3)
R Factor	(1.5)
Critical Power	(3.6)

* The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core.

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Bases Table B2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

THERMAL POWER	(3323) MW
Core Flow	(108.5) Mlb/hr
Dome Pressure	(1010.4) psig
Channel Flow Area	(0.1089) ft^2
R-Factor	High enrichment - (1.043) Medium enrichment - (1.039) Low enrichment - (1.030)

Bases Table B2.1.2-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
(1.525) to (1.575)	(2.1)
(1.475) to (1.525)	(8.9)
(1.425) to (1.475)	(10.5)
(1.375) to (1.425)	(3.1)
(1.325) to (1.375)	(5.2)
(1.275) to (1.325)	(2.1)
(1.225) to (1.275)	(5.2)
(1.175) to (1.225)	(2.1)
(1.125) to (1.175)	(6.3)
(1.075) to (1.125)	(5.8)
(1.025) to (1.075)	(2.1)
(0.975) to (1.025)	(46.6)
	100

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Bases Table B2.1.2-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

8x8 Rod Array

R-Factor High Enrichment	R-Factor Medium Enrichment	R-Factor Low Enrichment	Rod Sequence No.
(1.043)	(1.039)	(1.030)	1
(1.043)	(1.039)	(1.030)	2
(1.042)	(1.028)	(1.030)	3
(1.042)	(1.028)	(1.030)	4
(1.038)	(1.027)	(1.028)	5
(1.038)	(1.027)	(1.028)	6
(1.026)	(1.026)	(1.028)	7
≤(1.027)	≤(1.026)	≤(1.028)	8 thru 64

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2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum pressure transient of (110)%, (1375) psig, of design pressure, (1250) psig. The Safety Limit of (1325) psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to (1375) psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Piping Code, Section B31.1, which permits a maximum pressure transient of (120)%, (1380) psig, a design pressure, (1150) psig for suction piping and (1250) psig for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

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2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 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 exeeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section (15.1.12) of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at (1)% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to (21)% of RATED THERMAL POWER with the peak fuel enthalpy well below the licensing basis fuel failure design criterion of (170) cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of (15)% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all the possible sources of reactivity input, uniform control rod

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

withdrawal is the most probable cause of significant power increase. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than (5)% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The (15)% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-Upscale (120)% setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. (For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time delay is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics.) (A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.)

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when the design TOTAL PEAKING FACTOR is exceeded.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated a margin of ______ to the thermal hydraulic limit.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The results reported in Section 15 show that scram and isolation of all process lines, except main steam, at this level adequately protects the fuel and the pressure barrier, because MCHFR is greater than 1.0 in all cases, and system pressure does not reach the safety valve settings. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

6. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding. (No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.)

7. Primary Containment Pressure-High

High pressure in the drywell could indicate a break in the nuclear process systems. The reactor is tripped in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant. The trip setting was selected as low as possible without causing spurious trips.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of (10)% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed.

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection with or without coincidence with failure of the turbine bypass valve(s). The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than (30) milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a nominally 50% greater closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in Section (15.1.0) of the Final Safety Analysis Report.

11. Reactor Mode Switch in Shutdown Position

The reactor mode switch Shutdown position is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

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

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery or until the reactor is placed in an OPERATIONAL CONDITION in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL CONDITION or other specified applicability state shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION statements unless otherwise excepted. This provision shall not prevent passage through OPERATIONAL CONDITIONS required to comply with ACTION requirements.

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for individual Limiting Conditions for Operation unless otherwise stated in the individual Surveillance Requirements.

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

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

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

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SURVEILLANCE REQUIREMENTS (Continued)

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment.

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

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

a. During the time period:

1. From issuance of the Facility Operating License to the start of unit commercial operation, inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (*) Edition, and Addenda through _______* except where specific written relief has been granted by the Commission.

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

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

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

*Specific Code edition and addenda are to be specified consistent with 10 CFR 50.55a(b).

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1 The SHUTDOWN MARGIN shall be equal to or greater than:

a. (0.38)% delta k/k with the highest worth rod analytically determined, or

b. (0.28)% delta k/k with the highest worth rod determined by test.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

a. In OPERATIONAL CONDITION 1 or 2, reestablish the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4, immediately verify all control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS* and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.1.1 The SHUTDOWN MARGIN shall be determined to be equal to or greater than specified at any time during the fuel cycle:

a. By measurement, prior to or during the first startup after each refueling.

b. By measurement, within (500) MWD/T prior to the core average exposure at which the predicted SHUTDOWN MARGIN, including uncertainties and calculation biases, is equal to the limiting value.

c. Within one hour after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or is untrippable, except that the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod.

*Except movement of SRMs or special movable detectors.

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3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITION FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the reactivity different by more than 1% delta k/k, within 12 hours, either:

a. Perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected, or

b. Be in at least HOT SHUTDOWN. Determine and correct the cause and magnitude of the reactivity difference.

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% delta k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

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3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. POWER OPERATION may continue provided that the inoperable control rod:

      a) If withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

      b) Directional control valves are disarmed either:

         1) Electrically, or

         2) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

      Restore the inoperable control rod to OPERABLE status within 48 hours, or

   2. Be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods inoperable for causes other than addressed in ACTION a, above:

   1. The provisions of Specification 3.0.4 are not applicable and operation may continue provided that:

      a) Within one hour each inoperable withdrawn control rod is either:

         1) Verified to be separated from all other inoperable control rods by at lea t two control cells in all directions, or

         2) Fully inserted and then the directional control valves are disarmed either:

            (a) Electrically, or

            (b) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b) Within 2 hours, either:

1) The insertion capability of each inoperable withdrawn control rod is demonstrated by inserting the control rod at least one notch by drive water pressure within the normal operating range*, or

2) The inoperable control rod is fully inserted and then the directional control valves are disarmed either:

a) Electrically, or

b) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

c) Within one hour each fully inserted inoperable control rod directional control valves are disarmed either:

1) Electrically, or

2) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

2. Otherwise, be in at least HOT SHUTDOWN within 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 All withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch;

a. At least once per 7 days when above the preset power level of the RWM and RSCS, or

b. At least once per 24 hours when above the preset power level of the RWM and RSCS and three or more control rods are inoperable.

4.1.3.1.2 All control rods shall be demonstrated OPERABLE by performance of the Surveillance Requirements of Specifications 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

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CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. The provisions of Specification 3.0.4 are not applicable and operation may continue provided that;

   1. The control rod(s) with the slow insertion time is declared inoperable, and

   2. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds.

b. Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

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CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.3 The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
(46)	(0.375)
(36)	(1.096)
(26)	(2.000)
(6)	(4.000)

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.3 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Specification 4.1.3.2.

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FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.4 The average scram insertion time, from the fully withdrawn position, for the three fastest control rods in each group of four control rods arranged in a two-by-two array, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
(46)	(0.398)
(35)	(1.169)
(26)	(2.120)
(6)	(4.300)

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

a. The provisions of Specification 3.0.4 are not applicable and operation may continue provided that;

   1. The control rods with the slower than average scram insertion times are declared inoperable,

   2. An analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

   3. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 60 days when operation is continued with control rods in excess of the four control rod group scram insertion times.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.4 All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Specification 4.1.3.2.

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

   1. With one control rod scram accumulator inoperable:

      a) The provisions of Specification 3.0.4 are not applicable and operation may continue, provided that within 8 hours, either:

         1) The inoperable accumulator is restored to OPERABLE status, or

         2) The control rod associated with the inoperable accumulator is declared inoperable.

      b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

   2. With more than one control rod scram accumulator inoperable, either:

      a) Immediately verify control rod insertion capability by inserting at least one withdrawn control rod at least one notch by drive water pressure within the normal operating range and be in at least HOT SHUTDOWN within 6 hours, or

      b) Place the reactor mode switch in the Shutdown position.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod scram accumulator inoperable, fully insert the affected control rod and disarm the directional control valves either electrically or hydraulically, by closing the drive water withdraw and exhaust isolation valves within one hour. The provisions of Specification 3.0.3 are not applicable.

*At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the pressure and leak detectors are not in the alarmed condition,

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors to alarm at (970 to 940) psig on decreasing pressure.

2. Verifying that the accumulator pressure (and level) remains above the alarm set point(s) for greater than or equal to 20 minutes with no control rod drive pump operating.

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CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.6 All control rods shall be coupled to their drive mechanisms.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 and 2 with one control rod not coupled to its associated drive mechanism:

   1. The provisions of Specification 3.0.4 are not applicable and operation may continue provided that within 2 hours either:

      a) If permitted by the RWM and RSCS, the control rod drive mechanism is inserted to accomplish recoupling and recoupling is verified by withdrawing the control rod, and:

         1) Observing any indicated response of the nuclear instrumentation, and

         2) Demonstrating that the control rod will not go to the overtravel position.

      b) If recoupling is not accomplished on the first attempt or if not permitted by the RWM or RSCS, the control rod is declared inoperable and fully inserted, and the directional control valves are disarmed either:

         1) Electrically, or

         2) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours, either:

   1. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing the control rod and demonstrating that the control rod will not go to the overtravel position, or

   2. If recoupling is not accomplished, fully insert the control rod and disarm the directional control valve either:

      a) Electrically, or

      b) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

   3. The provisions of Specification 3.0.3 are not applicable.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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SURVEILLANCE REQUIREMENTS

4.1.3.6 A control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod does not go to the overtravel position;

a. Prior to reactor criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity,

b. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

c. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one or more control rod position indicators inoperable, except for the "Full-in" or "Full-out" indicators:

a) The provisions of Specification 3.0.4 are not applicable and operation may continue, provided that within one hour:

1) When THERMAL POWER is within the preset power level of the RSCS:

(a) The position of the control rod is determined by (an alternate method), or

(b) The control rod is moved to a position with an OPERABLE position indicator, or

(c) The control rod is declared inoperable.

2) When THERMAL POWER is greater than the preset power level of the RSCS:

(a) The position of the control rod is determined by (an alternate method), or

(b) The control rod is moved to a position with an OPERABLE position indicator, or

(c) The control rod is declared inoperable and is fully inserted and the directional control valves are disarmed either:

(1) Electrically, or

(2) Hydraulically by closing the drive water withdraw and exhaust isolation valves.

b) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. With one or more control rod "Full-in" and "Full-out" position indicators inoperable:

   a) The affected control rod may be bypassed in the Rod Sequence Control System, the provisions of Specification 3.0.4 are not applicable and operation may continue, provided that:

      1) The actual control rod position is known, and

      2) The affected control rod is moved in the correct sequence and pattern.

   b) Otherwise, be in at least HOT SHUTDOWN within 12 hours.

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or fully insert the control rod. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.7.1 The control rod position indication system shall be determined OPERABLE by verifying:

a. At least once per 24 hours that the position of each control rod is indicated,

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1, and

c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6.b.

4.1.3.7.2 When the RSCS is required to be OPERABLE, the position and bypassing of control rods with an inoperable "Full-in" or Full-out" position indicator shall be verified by a second licensed operator or other technically qualified member of the unit technical staff.

*At least each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.8 The control rod drive housing support shall be in place.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.8 The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

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3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*, when THERMAL POWER is less than or equal to (20)% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, the provisions of Specification 3.0.4 are not applicable, operation may continue and control rod movement is permitted provided that a second licensed operator or other technically qualified member of the unit technical staff is present at the reactor control console and verifies compliance with the prescribed control rod pattern. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.1.4.1.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper annunication of the selection error of at least one out-of-sequence control rod (in each distinct RWM group).

b. In OPERATIONAL CONDITION 2 prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by attempting to withdraw an out-of-sequence control rod (one) notch.

c. In OPERATIONAL CONDITION 1 within one hour after RWM automatic initiation when reducing THERMAL POWER, by verifying the rod block function by attempting to withdraw an out-of-sequence control rod (one) notch.

d. By verifying the control rod patterns and sequence input to the RWM computer is correctly loaded following any loading of the program into the computer.

*Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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ROD SEQUENCE CONTROL SYSTEM (Group Notch Type)

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod sequence control system (RSCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to (20)% RATED THERMAL POWER.

ACTION:

With the RSCS inoperable:

a. Control rod withdrawal for reactor startup shall not begin.

b. If control rod withdrawal has started and THERMAL POWER is less than or equal to (20)% of RATED THERMAL POWER, control rod movement shall not be permitted, except by scram.

c. With THERMAL POWER being reduced by control rod insertion, do not continue control rod insertion except by a scram.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. Selecting and attempting to move an inhibited control rod:

   1. After withdrawal of the first insequence control rod for each reactor startup, and

   2. Within one hour after rod inhibit mode automatic initiation when reducing THERMAL POWER.

b. Attempting to move a control rod more than one notch as soon as the group notch mode is automatically initiated during:

   1. Control rod withdrawal for each reactor startup, and
   2. Power reduction.

c. Performance of the comparator check of the group notch circuits prior to control rod:

   1. Withdrawal for each reactor startup, and
   2. Insertion to reduce THERMAL POWER to less than 20% of RATED THERMAL POWER.

---

*See Special Test Exception 3.10.2

#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

ROD SEQUENCE CONTROL SYSTEM (Banked Position Type)

LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod sequence control system (RSCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to (20)% RATED THERMAL POWER.

ACTION:

With the RSCS inoperable:

a. Control rod withdrawal for reactor startup shall not begin.

b. If control rod withdrawal has started and THERMAL POWER is less than or equal to (20)% of RATED THERMAL POWER, control rod movement shall not be permitted, except by scram.

c. With THERMAL POWER being reduced by control rod insertion, do not continue control rod insertion except by a scram.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

a. Performance of the self-test prior to:

   1. Each reactor startup, and

   2. Rod inhibit mode automatic initiation when reducing THERMAL POWER.

b. Attempting to select and move an inhibited control rod:

   1. After withdrawal of the first insequence control rod for each reactor startup, and

   2. Within one hour after rod inhibit mode automatic initiation when reducing THERMAL POWER.

*See Special Test Exception 3.10.2
#Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RSCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both Rod Block Monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than the preset power level of the RWM and RSCS.

ACTION:

a. With one RBM channel inoperable, POWER OPERATION may continue provided that the reactor is not operating on a LIMITING CONTROL ROD PATTERN and the inoperable RBM channel is restored to OPERABLE status within 24 hours; otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and during the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod (withdrawal) when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE with:

a. An OPERABLE flow path from the storage tank to the reactor core containing two pumps and two inline explosive injection valves,

b. The contained solution concentration and temperature within the limits of Figure 3.1.5-1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

2. With the standby liquid control system inoperable, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 5*:

1. With one pump and/or one explosive valve inoperable, restore the inoperable pump and/or explosive valve to OPERABLE status within 30 days or fully insert all insertable control rods within the next hour.

2. With the standby liquid control system inoperable, fully insert all insertable control rods within one hour.

3. The provisions of Specification 3.0.3 are not applicable.

*With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that;

1. The temperature of the sodium pentaborate solution is within the limits of Figures 3.1.5-1,

2. The volume of sodium pentaborate is greater than or equal to ( ) gallons, and,

3. The heat tracing circuit is OPERABLE by determining the temperature of the (pump suction piping) to be greater than or equal to (70)°F.

b. At least once per 31 days by;

1. Starting each pump and recirculating demineralized water to the test tank.

2. Verifying the continuity of the explosive charge.

3. Determining that the weight of sodium pentaborate is greater than or equal to ( ) lbs and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

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

c. At least once per 18 months during shutdown by;

1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired. Both injection loops shall be tested in 36 months.

2. Demonstrating that the minimum flow requirement of (41.2) gpm at a pressure of (1220) psig is met.

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limit of Figure 3.1.5-1.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Demonstrating that the pump relief valve setpoint is greater than or equal to (1600) psig and verifying that the relief valve does not actuate during recirculation to the test tank.

4. **Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by (pumping from the storage tank to the test tank).

**This test shall also be performed whenever both heat tracing circuits have been found to be inoperable and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.

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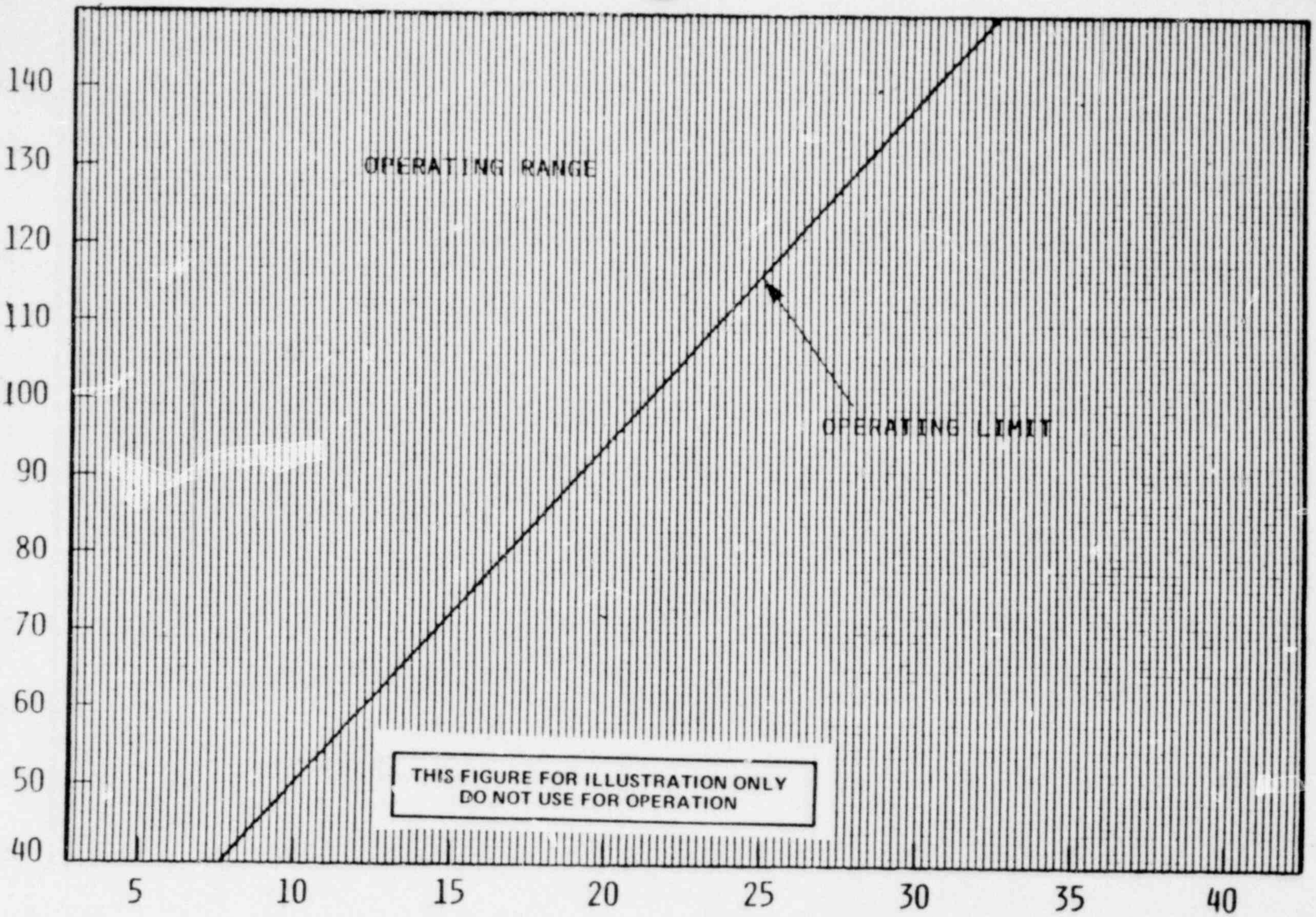

Concentration, % by weight

SODIUM PENTABORATE SOLUTION
TEMPERATURE/CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figure 3.2.1-1, 3.2.1-2, and 3.2.1-3..

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With a APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figure 3.2.1-1, 3.2.1-2, and 3.2.1-3:

a. At least once per 24 hours,

b. Within ( ) hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

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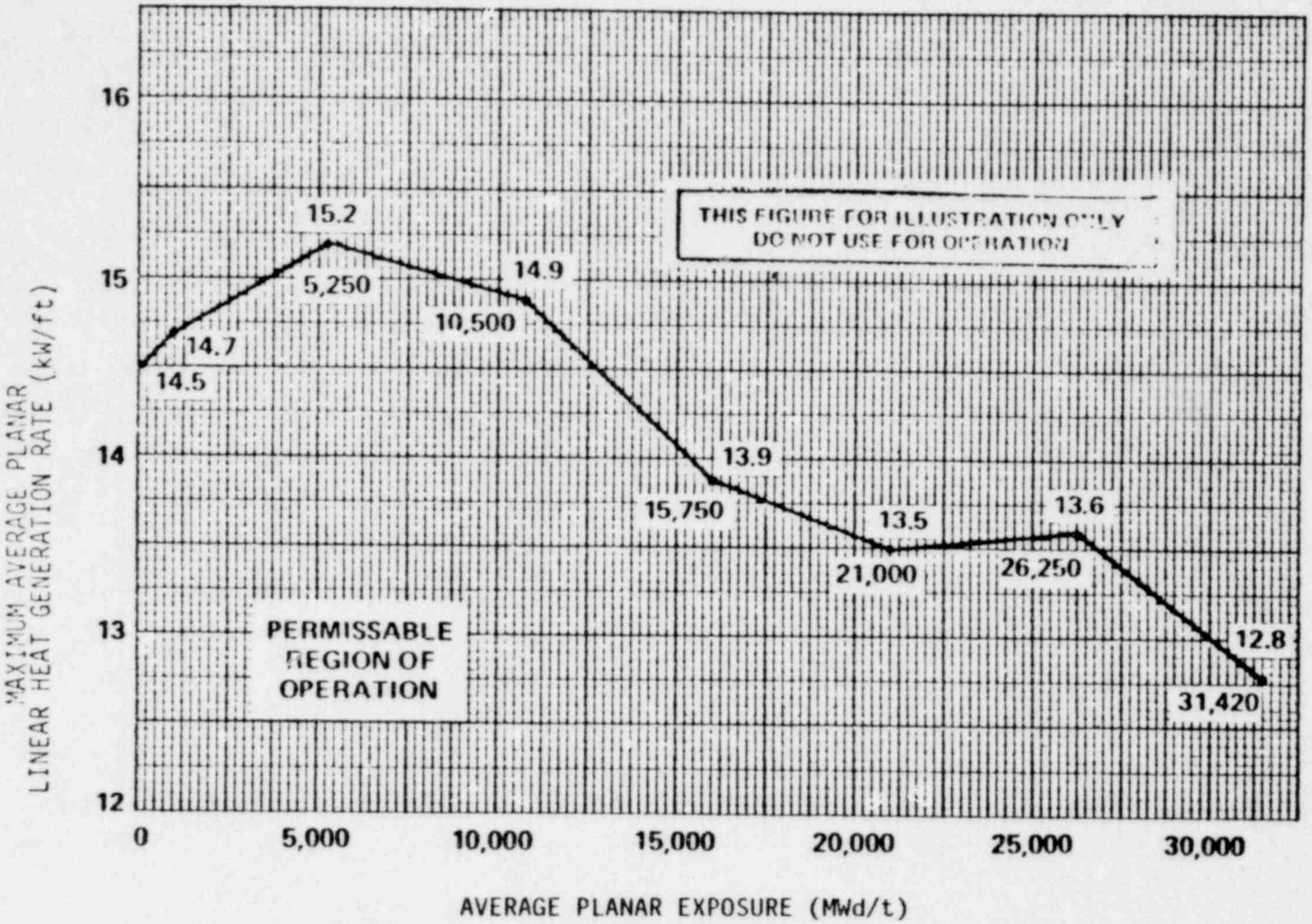

MAXIMUM AVERAGE PLANNAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES ( )
Figure 3.2.1-1

POOR ORIGINAL —

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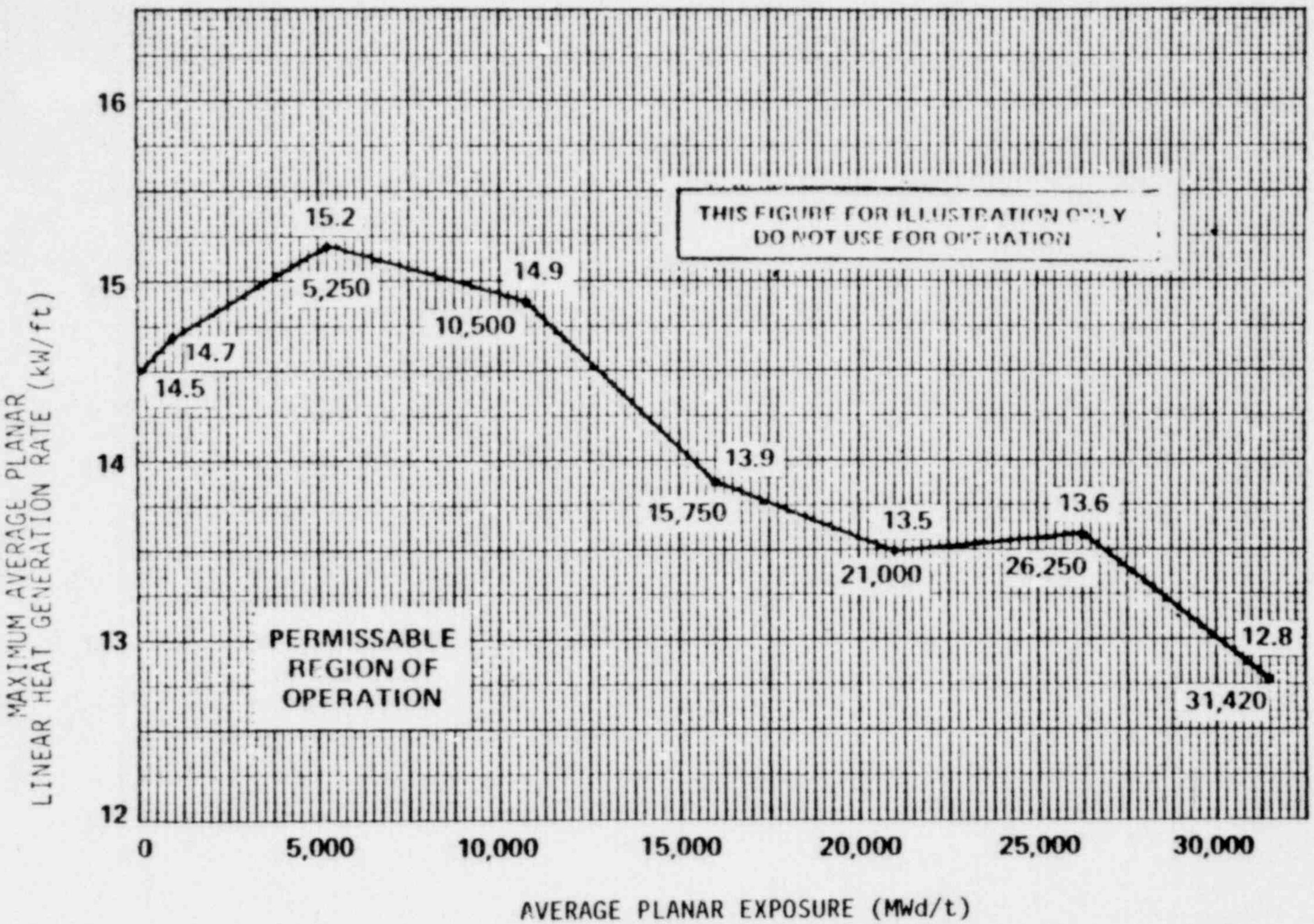
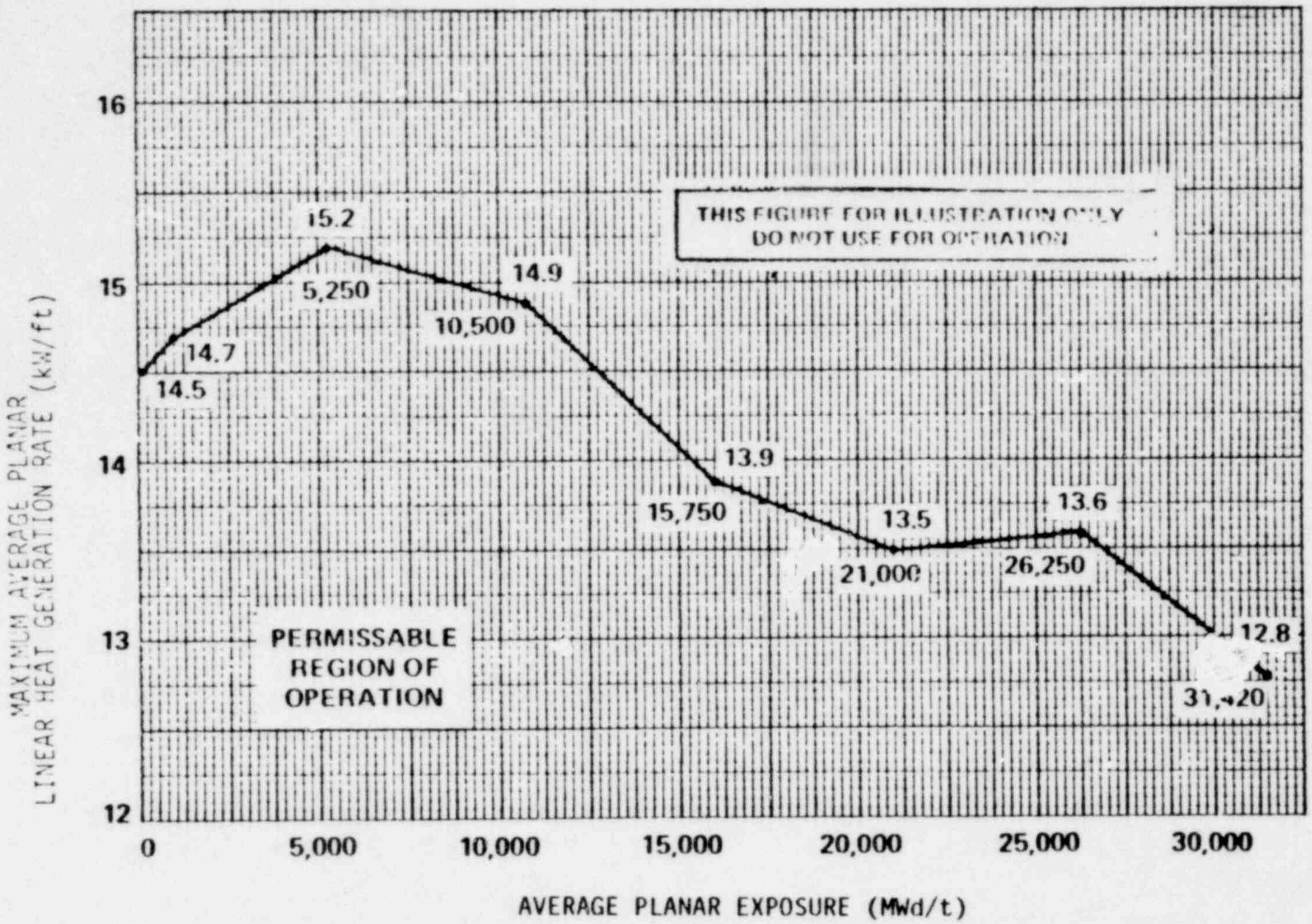


MAXIMUM AVERAGE PLANNAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURES
INITIAL CORE FUEL TYPES (   )
Figure 3.2.1-2



MAXIMUM AVERAGE PLANNAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPES (   )
Figure 3.2.1-3

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased simulated thermal power-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

$$S \leq (0.66W + (54)\%)\ T$$
$$S_{RB} \leq (0.66W + (42)\%)\ T$$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow in percent of rated flow,
T = Lowest value of the ratio of design TPF divided by the MTPF obtained for any class of fuel in the core, T greater than or equal to 1.0, and
Design TPF for 8 x 8 fuel = (2.43).

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint or the flow biased simulated thermal power-upscale control rod block trip setpoint less conservative than S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and S_{RB} to within the required limits within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The MTPF for each class of fuel shall be determined, the value of T calculated, and the flow biased scram and control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within ( ) hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with MTPF greater than or equal to (2.43).

3/4.2.3 MINIMUM CRITICAL POWER RATIO

POOR ORIGINAL

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR), as a function of core flow, shall be equal to or greater than MCPR times the K_f shown in Figure 3.2.3-1, provided that the end-of-cycle recirculation pump trip system is OPERABLE per Specification 3.3.4.2, with:

a. MCPR for 7 x 7 fuel = (1.20*).
b. MCPR for 8 x 8 fuel = (1.20).

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2 from:

   1. Beginning-of-cycle (BOC) to end-of-cycle (EOC) minus (2000) MWD/t, operation may continue and the provisions of Specification 3.0.4 are not applicable.

   2. EOC minus (2000) MWD/t to EOC, within one hour determine that MCPR, as a function of core flow, is equal to or greater than MCPR times the K_f shown in Figure 3.2.3-1 with:

      a) MCPR for 7x7 fuel = (1.21*), and
      b) MCPR for 8x8 and 8x8R fuel = (1.27).

b. With MCPR, as a function of core flow, less than the applicable limit determined from Figure 3.2.3-1, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR, as a function of core flow, shall be determined to be equal to or greater than the applicable limit determined from Figure 3.2.3-1:

a. At least once per 24 hours,

b. Within ( ) hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

*For 7x7 fuel assemblies, the k_f factor is based on the (112)% flow curve of Figure 3.2.3-1 rather than on the actual setpoint of (102.5)%.

SURVEILLANCE REQUIREMENTS (Continued)

(4.2.3.2 Each reactor coolant system recirculation pump MG set mechanical and electrical overspeed stop shall be demonstrated OPERABLE at least once per 18 months by verifying that each overspeed stop trips the associated pump at a set point of less than or equal to.

a. ( )% of rated core flow for the mechanical overspeed stop, and

b. ( )% of rated core flow for the electrical overspeed stop.)

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Core Flow, % of Rated Core Flow

K_f FACTOR

Figure 3.2.3-1

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3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed (13.4) kw/ft.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less then (25)% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

b. Within ( ) hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

* If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - Upscale	2, $5^{(b)}$	3	1
	3, 4	2	2
b. Inoperative	2, $5^{(b)}$	3	1
	3, 4	2	2
2. Average Power Range Monitor:			
a. Neutron Flux - Upscale	2, $5^{(b)}$	2	1
	3, 4	2	2
b. Flow Biased Simulated Thermal Power - Upscale	1	2	3
c. Neutron Flux - Upscale	1	2	3
d. Inoperative	1, 2, $5^{(b)}$	2	4
e. LPRM	1, 2, 5	(c)	NA
3. Reactor Vessel Steam Dome Pressure - High	1, $2^{(d)}$	2	5
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	5
5. Main Steam Line Isolation Valve - Closure	$1^{(e)}$	4	3
6. Main Steam Line Radiation - High	1, $2^{(d)}$	2	6
7. Primary Containment Pressure - High	1, $2^{(f)}$	2	5

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5$^{(g)}$	2	4
9. Turbine Stop Valve - Closure	1$^{(h)}$	4	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1$^{(h)}$	2	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors.

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry during CORE ALTERATIONS and during shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) These functions are automatically bypassed when turbine first stage pressure is $\leq$ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale	NA
b. Flow Biased Simulated Thermal Power - Upscale	$\leq$ (0.09**)
c. Fixed Neutron Flux - Upscale	$\leq$ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ (1.05)
5. Main Steam Line Isolation Valve - Closure	$\leq$ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1978. See Regulatory Guide 1.18, November 1977.)

(**Not including simulated thermal power time constant.)

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - Upscale	S/U[(c)], S	S/U[(b)]	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:				
a. Neutron Flux - Upscale	S/U[(c)], S	S/U[(b)], W	SA	2
	S	W	SA	3, 4, 5
b. Flow Biased Simulated Thermal Power - Upscale	S	S/U[(b)], W	W[(d)(e)], SA	1
c. Neutron Flux - Upscale	S	S/U[(b)], W	W[(d)], SA	1
d. Inoperative	NA	W	NA	1, 2, 5
e. LPRM	S	NA	(f)	1, 2, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	M	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	W[(g)]	R[(h)]	1, 2
7. Primary Containment Pressure - High	NA	M	Q	1, 2

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TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

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

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) The IRM and SRM channels shall be determined to overlap for at least (   ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least (   ) decades during each controlled shutdown, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.

(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Instrument alignment using the installed standard current source.

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

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3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition* within one hour.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken,

**If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

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SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation function.

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TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	(2, 6, 8)(c)	2	1, 2, 3	20
2) Low Low, Level 2	(1, 3)(d)	2	1, 2, 3	20
b. Drywell Pressure - High	(2, 6)(c)(d)	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	(1, 7)(d)(e)	2	1, 2, 3	21
2) Pressure - Low	(1)	2	1	22
3) Flow - High	(1)(d)	2/line	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	(1)	2/line	1, 2, 3	21
e. Main Steam Line Tunnel Δ Temperature - High	(1)	2	1, 2, 3	21
f. Condenser Vacuum - Low	(1)	2	1, 2*, 3*	21
g. Manual Initiation	(1, 2, 3, 6, 7, 8)	(1)	1, 2, 3	23
h. ________	____	___	______	___
2. SECONDARY CONTAINMENT ISOLATION				
a. Plant Exhaust Plenum Radiation - High	(6)(c)(f)	2(g)	1, 2, 3, 5 and **	24
b. Drywell Pressure - High	(6)(c)(d)(f)	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low, Level 3	(6)(c)(f)	2	1, 2, 3, 5 and **	24
d. Refueling Floor Exhaust Radiation - High	(6)(c)(d)(f)	2(g)	1, 2, 3, 5 and **	24
e. Manual Initiation	(6)	(1)	1, 2, 3, 5 and **	24
f. ________	___	___	______	___

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	(3)	1	1, 2, 3	25
b. Area Temperature - High	(3)	(2)	1, 2, 3	25
c. SLCS Initiation	(3)	NA	1, 2, 3	25
d. Area Ventilation Δ Temp. - High	(3)	(2)	1, 2, 3	25
e. Reactor Vessel Water Level - Low Low, Level 2	(3)	2	1, 2, 3	25
f. Manual Initiation	(3)	(1)	1, 2, 3	25
g. ________________	___	___	______	___
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	(4)	1	1, 2, 3	25
b. RCIC Steam Supply Pressure - Low	(4)	2	1, 2, 3	25
c. RCIC Turbine Exhaust Diaphragm Pressure - High	(4)	2	1, 2, 3	25
d. RCIC Equipment Room Temperature - High	(4)	1	1, 2, 3	25
e. RCIC Steam Line Tunnel Temperature - High	(4)	1	1, 2, 3	25
f. RCIC Steam Line Tunnel Δ Temperature - High	(4)	1	1, 2, 3	25
g. Manual Initiation	(4)	(1)	1, 2, 3	25
h. ____________________	-	-	-	-

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	(5)	___	1, 2, 3, 4, 5	25
b. RHR Area Cooler Temperature - High	(5)	___	1, 2, 3, 4, 5	25
c. RHR Flow - High	(5)	___	1, 2, 3, 4, 5	25
d. Manual Initiation	(5)	(1)_	1, 2, 3, 4, 5	25
e. ____________________	-	-	-	-
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	(5)	-	1, 2, 3, 4, 5	26
b. RHR Cut-in Permissive Pressure - High	(5)	-	1, 2, 3, 4	26
c. Manual Initiation	(5)	(1)	1, 2, 3, 4, 5	26
d. _______________	-	-	-	-

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 2 hours or be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours.

ACTION 22 - Be in at least STARTUP within 2 hours.

ACTION 23 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 25 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 26 - Lock the affected system isolation valves closed within one hour and declare the affect system inoperable.

NOTES

* May be bypassed with reactor steam pressure $\leq$ (1043) psig and all turbine stop valves closed.

** When handling irradiated fuel or a spent fuel shipping cask in the secondary containment.

(a) See Specification 3.6.3.1, Table 3.6.3.1-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the (isolation) mode of operation.

(e) Also trips the mechanical vacuum pumps.

(f) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(g) Two upscale and/or downscale actuate the trip system.

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TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ (12.5) inches*	≥ (11.0) inches*
2) Low Low, Level 2	≥ -(38) inches*	≥ -( ) inches*
b. Drywell Pressure - High	≤ (1.69) psig	≤ (1.89) psig
c. Main Steam Line		
1) Radiation - High	≤ (3.0) x full power background	≤ (3.6) x full power background
2) Pressure - Low	≥ (854) psig	≥ (834) psig
3) Flow - High	≤ (140)% of rated flow	≤ (145)% of rated flow
d. Main Steam Line Tunnel Temperature - High	≤ (140)°F	≤ ( )°F
e. Main Steam Line Tunnel Δ Temperature - High	≤ (50)°F	≤ ( )°F
f. Condenser Vacuum - Low	≥ (23) inches Hg Absolute Pressure	≥ ( ) inches Hg absolute pressure
g. Manual Initiation	NA	NA
h. ________________	—	—
2. SECONDARY CONTAINMENT ISOLATION		
a. Plant Exhaust Plenum Radiation - High	≤ (4.5) mr/hr**	≤ ( ) mr/hr**
b. Drywell Pressure - High	≤ (1.69) psig	≤ (1.89) psig
c. Reactor Vessel Water Level - Low, Level 3	≥ (12.5) inches*	≥ (11.0) inches*
d. Refueling Floor Exhaust Radiation - High	≤ (35) mr/hr**	≤ ( ) mr/hr**
e. Manual Initiation	NA	NA
f. ________________	—	—

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
3.	REACTOR WATER CLEANUP SYSTEM ISOLATION		
a.	Δ Flow - High	≤ (70)gpm	≤ (87.5) gpm
b.	Area Temperature - High	≤ (140)°F	≤ ( )°F
c.	SLCS Initiation	NA	NA
d.	Area Ventilation Δ Temperature - High	≤ (50)°F	≤ ( )°F
e.	Reactor Vessel Water Level - Low Low, Level 2	≥ -(38) inches*	≥ -( ) inches*
f.	Manual Initiation	NA	NA
g.	____________________	__________	__________
4.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a.	RCIC Steam Line Flow - High	≤ (300)% of rated flow	≤ ( )% of rated flow
b.	RCIC Steam Supply Pressure - Low	≥ (60) psig	≥ (50) psig
c.	RCIC Turbine Exhaust Diaphragm Pressure - High	≤ (10) psig	≤ (20) psig
d.	RCIC Equipment Room Temperature - High	≥ (175)°F ≤ (225)°F	≥ ( )°F ≤ ( )°F
e.	RCIC Steam Line Tunnel Temperature - High	≤ (200)°F	≤ ( )°F
f.	RCIC Steam Line Tunnel Δ Temperature - High	≤ (100)°F	≤ ( )°F
g.	Manual Initiation	NA	NA
h.	____________________	__________	__________

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION		
a. RHR Equipment Area Δ Temperature - High	$\leq$ (100)°F**	$\leq$ ( )°F**
b. RHR Area Cooler Temperature - High	$\leq$ (200)°F**	$\leq$ ( )°F**
c. RHR Flow - High	$\leq$ (300)% of rated flow	$\leq$ ( )% of rated flow
d. Manual Initiation	NA	NA
e. ________________	________	________
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level - Low, Level 3	$\geq$ 12.5 inches*	$\geq$ ( ) inches*
b. RHR Cut-in Permissive Pressure - High	$\leq$ (190) psig***	$\leq$ ( ) psig***
c. Manual Initiation	NA	NA
d. ________________	___	___

*See Bases Figure B 3/4 3-1.

**Initial setpoint. Final setpoint to be determined during startup testing.

***Corrected for cold water head with reactor vessel flooded.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Level 3	≤ (13)(a)
2) Low Low, Level 2	≤ (1.0)*/≤ (13)(a)
b. Drywell Pressure - High	≤ (13)(a)
c. Main Steam Line	
1) Radiation - High(b)	≤ (1.0)*/≤ (13)
2) Pressure - Low	≤ (1.0)*/≤ (13)
3) Flow - High	≤ (0.5)*/≤ (13)
d. Main Steam Line Tunnel Temperature - High	(NA)
e. Main Steam Line Tunnel Δ Temperature - High	(NA)
f. Condenser Vacuum - Low	(NA)
g. Manual Initiation	NA
h. ______________	____
2. SECONDARY CONTAINMENT ISOLATION	
a. Plant Exhaust Plenum Radiation - High(b)	≤ (13)(a)
b. Drywell Pressure - High	≤ (13)(a)
c. Reactor Vessel Water Level - Low, Level 3	≤ (13)(a)
d. Refueling Floor Exhaust Radiation - High(b)	≤ (13)(a)
e. Manual Initiation	NA
f. ______________	____
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤ (13)(a)
b. Area Temperature - High	(NA)
c. SLCS Initiation	(NA)
d. Area Ventilation Temperature ΔT - High	(NA)
e. Reactor Vessel Water Level - Low Low, Level 2	≤ (13)(a)
f. Manual Initiation	NA
g. ______________	____

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TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
4.	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a.	RCIC Steam Line Flow - High	≤ (13)(a)
b.	RCIC Steam Supply Pressure - Low	(NA)
c.	RCIC Turbine Exhaust Diaphragm Pressure - High	(NA)
d.	RCIC Equipment Room Temperature - High	(NA)
e.	RCIC Steam Line Tunnel Temperature - High	(NA)
f.	RCIC Steam Line Tunnel Δ Temperature - High	(NA)
g.	Manual Initiation	NA
h.	________________	___
5.	RHR SYSTEM STEAM CONDENSING MODE ISOLATION	
a.	RHR Equipment Area Δ Temperature - High	(NA)
b.	RHR Area Cooler Temperature - High	(NA)
c.	RHR Flow - High	(NA)
d.	Manual Initiation	NA
e.	___________________________	______
6.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a.	Reactor Vessel Water Level - Low, Level 3	≤ (13)(a)
b.	RHR Cut-in Permissive Pressure - High	(NA)
c.	Manual Initiation	NA
d.	____________________	___

(a)The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes the delay for diesel generator starting assumed in the accident analysis.

(b)Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time only. No diesel generator delays assumed for (_______) valves.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

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TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	S	M	R	1, 2, 3
2) Low Low, Level 2	S	M	R	1, 2, 3
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	M(a)	R(b)	1, 2, 3
2) Pressure - Low	NA	M	Q	1
3) Flow - High	S	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
e. Main Steam Line Tunnel Δ Temperature - High	NA	M	Q	1, 2, 3
f. Condenser Vacuum - Low	NA	M	Q	1, 2*, 3*
g. Manual Initiation	NA	M(c)	NA	1, 2, 3
h. ________	—	—	—	________
2. SECONDARY CONTAINMENT ISOLATION				
a. Plant Exhaust Plenum Radiation - High	S	M(a)	R	1, 2, 3, 5 and**
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3, 5 and**
d. Refueling Floor Exhaust Radiation - High	S	M(a)	R	1, 2, 3, 5 and**
e. Manual Initiation	NA	M(c)	NA	1, 2, 3, 5 and**
f. ________	—	—	—	________

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TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Area Temperature - High	NA	M	Q	1, 2, 3
c. SLCS Initiation	NA	M(d)	NA	1, 2, 3
d. Area Ventilation Δ Temperature - High	NA	M	Q	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
f. Manual Initiation	NA	M(c)	NA	1, 2, 3
g. ________________	_	_	_	1, 2, 3
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	NA	M	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	M	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	M	Q	1, 2, 3
e. RCIC Steam Line Tunnel Temperature - High	NA	M	Q	1, 2, 3
f. RCIC Steam Line Tunnel Δ Temperature - High	NA	M	Q	1, 2, 3
g. Manual Initiation	NA	M(c)	NA	1, 2, 3
h. ________________________	_	_	_	________

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TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	NA	M	Q	1, 2, 3, 4, 5
b. RHR Area Cooler Temperature - High	NA	M	Q	1, 2, 3, 4, 5
c. RHR Flow - High	NA	M	Q	1, 2, 3, 4, 5
d. Manual Initiation	NA	M(c)	NA	1, 2, 3, 4, 5
e. ______	—	—	—	______
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3, 4, 5
b. RHR Cut-in Permissive Pressure - High	NA	M	Q	1, 2, 3, 4
c Manual Initiation	NA	M(c)	NA	1, 2, 3, 4, 5
d. ______	—	—	—	______

* When reactor steam pressure ≥ ( ) psig and/or any turbine stop valve is open.

** When handling irradiated fuel or a spent fuel shipping cask in the secondary containment.

(a) Instrument alignment using the installed standard current source.

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

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

(d) Each train or logic channel shall be tested at least every other 31 days.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS function.

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TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. LOW PRESSURE CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. Injection Valve Differential Pressure - Low (Permissive)	1	1, 2, 3, 4*, 5*	31
d. Pump Discharge Pressure - High (Bypass)	1	1, 2, 3, 4*, 5*	31
e. Reactor Vessel Pressure - Low (Permissive)	2	1, 2, 3, 4*, 5*	30
f. Bus Power Monitor	1(c)	1, 2, 3, 4*, 5*	32
g. Initiation Logic	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1	1, 2, 3, 4*, 5*	33
i. ________________	___	____________	__
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. Injection Valve Differential Pressure - Low (Permissive)	1/loop	1, 2, 3, 4*, 5*	31
d. RHR Pump Start - Auto Sequence Timer	1/pump	1, 2, 3, 4*, 5*	31
e. Pump Discharge Pressure - High (Bypass)	1/pump	1, 2, 3, 4*, 5*	31
f. Bus Power Monitor	1/bus(c)	1, 2, 3, 4*, 5*	32
g. Initiation Logic	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1	1, 2, 3, 4*, 5*	33
i. ________________	___	____________	__
3. HIGH PRESSURE CORE SPRAY SYSTEM#			
a. Reactor Vessel Water Level - Low Low, Level 2	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. Condensate Storage Tank Level - Low	2(d)	1, 2, 3, 4*, 5*	34
d. Suppression Chamber Water Level - High	2(d)	1, 2, 3, 4*, 5*	34
e. Reactor Vessel Water Level - High	2(e)	1, 2, 3, 4*, 5*	31
f. Pump Discharge Pressure - High (Bypass)	1	1, 2, 3, 4*, 5*	31
g. Pump Suction Pressure - Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. HPCS System Flow Rate - Low (Permissive)	1	1, 2, 3, 4*, 5*	31
i. Bus Power Monitor	1(c)	1, 2, 3, 4*, 5*	32
j. Initiation Logic	1	1, 2, 3, 4*, 5*	31
k. Manual Initiation	1	1, 2, 3, 4*, 5*	33
l. ________________	___	____________	__

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM[#]			
a. Reactor Vessel Water Level - Low Low Low, Level 1, coincident with	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Low Pressure Core Spray Pump Discharge Pressure - High (Permissive)	1	1, 2, 3	31
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	1/loop	1, 2, 3	31
f. Reactor Vessel Water Level - Low, Level 3 (Permissive)	2	1, 2, 3	30
g. Bus Power Monitor	1/bus[(c)]	1, 2, 3	32
h. Initiation Logic	1	1, 2, 3	31
i. Manual Initiation	1/(valve)	1, 2, 3	33
j. ________	________	________	—

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the (associated) emergency diesel generators.
(c) Alarm only.
(d) One trip system. Provides signal to HPCS pump suction valves only.
(e) On 2 out of 2 logic, provides signal to close HPCS pump discharge valve only.
* When the system is required to be OPERABLE per Specification 3.5.1, 3.5.3.1 or 3.5.3.2.
\# Not required to be OPERABLE when reactor steam dome pressure is $\leq$ (113) psig.

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel in the tripped condition within one hour or declare the associated ECCS inoperable.

b. For both trip systems, declare the associated ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the minimum OPERABLE Channels per Trip System requirement, declare the associated ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, verify bus power availability at least once per 12 hours or declare the associated ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the associated ECCS or ADS valve inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the HPCS system inoperable.

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TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. LOW PRESSURE CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -(146) inches*	≥ -( ) inches*
b. Drywell Pressure - High	≤ (1.69) psig	≤ (1.89) psig
c. Injection Valve Differential Pressure-Low	(729) ± ( ) psid	(709) ± ( ) psid
d. Pump Discharge Pressure-High	≥ ( ) gpm	≥ ( ) gpm
e. Reactor Vessel Pressure-Low	≥ (150) psig, (increasing)	≥ ( ) psig, (increasing)
f. Bus Power Monitor	≥ ( ) volts	≥ ( ) volts
g. Initiation Logic	NA	NA
h. Manual Initiation	NA	NA
i. ____ ________ __	—	—
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -(146) inches*	≥ -( ) inches*
b. Drywell Pressure - High	≤ (1.69) psig, (increasing)	≤ (1.89) psig, (increasing)
c. Injection Valve Differential Pressure - Low	≥ (729) psid, (decreasing)	≥ (709) psid, (decreasing)
d. RHR Pump Start - Auto Sequence Timers	≤ (5) seconds	≤ ( ) seconds
e. Pump Discharge Pressure-High	≥ ( ) psig	≥ ( ) psig
f. Bus Power Monitor	≥ ( ) volts	≥ ( ) volts
g. Initiation Logic	NA	NA
h. Manual Initiation	NA	NA
i. ________________	—	—
3. HIGH PRESSURE CORE SPRAY SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -(38) inches*	≤ -( ) inches*
b. Drywell Pressure - High	≤ (1.69) psig	≤ (1.89) psig
c. Condensate Storage Tank Level - Low	≥ (0) inches	≥ ( ) inches
d. Suppression Chamber Water Level - High	≤ (2) inches	≤ ( ) inches
e. Reactor Vessel Water Level - High	≤ (55.5) inches*	≤ ( ) inches*
f. Pump Discharge Pressure - High	≥ (120) psig	≥ ( ) psig
g. Pump Suction Pressure - Low	≥ (4) ≤ (100) psig	≥ ( ) ≤ ( ) psig
h. HPCS System Flow Rate - Low	≥ (470) gpm	≥ ( ) gpm
i. Bus Power Monitor	≥ ( ) volts	≥ ( ) volts
j. Initiation Logic	NA	NA
k. Manual Intiation	NA	NA
l. ________________	—	—

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Reactor Water Level - Low Low Low, Level 1	$\geq$ -(14[illegible]) inches*	$\geq$ -( ) inches*
b. Drywell Pressure-High	$\leq$ (1.69) psig	$\leq$ (1.89) psig
c. ADS Timer	$\geq$ (90), $\leq$ (120) seconds	$\geq$ (90), $\leq$ ( ) seconds
d. Low Pressure Core Spray Pump Discharge Pressure - High	$\geq$ (150) psig, (increasing)	$\geq$ ( ) psig, (increasing)
e. RHR LPCI Mode Pump Discharge Pressure - High	$\geq$ (150) psig, (increasing)	$\geq$ ( ) psig, (increasing)
f. Reactor Vessel Water Level-Low, Level 3	$\geq$ (12.5) inches*	$\geq$ ( ) inches*
g. Bus Power Monitor	$\geq$ ( ) volts	$\geq$ ( ) volts
h. Initiation Logic	NA	NA
i. Manual Initiation	NA	NA
j. ________________	__	__

*See Bases Figure B 3/4 3-1.

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TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	$\leq$ (40)
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	$\leq$ (40)
3. HIGH PRESSURE CORE SPRAY SYSTEM	$\leq$ (27)
4. AUTOMATIC DEPRESSURIZATION SYSTEM	NA

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. LOW PRESSURE CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	Q	1, 2, 3
c. Injection Valve Differential Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
d. Pump Discharge Pressure-High	NA	M	Q	1, 2, 3, 4*, 5*
e. Reactor Vessel Pressure - Low	NA	M	Q	1, 2, 3, 4*, 5*
f. Bus Power Monitor	S	M	R	1, 2, 3, 4*, 5*
g. Initiation Logic	NA	M(a)	NA	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	M(a)	NA	1, 2, 3, 4*, 5*
i. ________	__	__	__	________
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	M	O	1, 2, 3
c. Injection Valve Differential Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
d. RHR Pump Start-Auto Sequence Timers	NA	M(b)	R	1, 2, 3, 4*, 5*
e. Pump Discharge Pressure-High	NA	M	Q	1, 2, 3, 4*, 5*
f. Bus Power Monitor	S	M	R	1, 2, 3, 4*, 5*
g. Initiation Logic	NA	M(a)	NA	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	M(a)	NA	1, 2, 3, 4*, 5*
i. ________	__	__	__	________

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. HIGH PRESSURE CORE SPRAY SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	Q	1, 2, 3, 4*, 5*
d. Suppression Chamber Water Level - High	S	M	R	1, 2, 3, 4*, 5*
e. Reactor Vessel Water Level-High	NA	M	Q	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	NA	M	Q	1, 2, 3, 4*, 5*
g. Pump Suction Pressure-Low	NA	M	Q	1, 2, 3, 4*, 5*
h. HPCS System Flow Rate-Low	NA	M	Q	1, 2, 3, 4*, 5*
i. Bus Power Monitor	S	M	R	1, 2, 3, 4*, 5*
j. Initiation Logic	NA	M	NA	1, 2, 3, 4*, 5*
k. Manual Initiation	NA	M(a)	NA	1, 2, 3, 4*, 5*
l. ________	—	—	—	________
4. AUTOMATIC DEPRESSURIZATION SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Drywell Pressure-High	NA	M	Q	1, 2, 3
c. ADS Timer	NA	M(b)	R	1, 2, 3
d. Low Pressure Core Spray Pump Discharge Pressure-High	NA	M	Q	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure-High	S	M	R	1, 2, 3
f. Reactor Vessel Water Level-Low Level 3	S	M	R	1, 2, 3
g. Bus Power Monitor	S	M	R	1, 2, 3
h. Initiation Logic	NA	M	NA	1, 2, 3
i. Manual Initiation	NA	M(a)	NA	1, 2, 3
j. ________	—	—	—	________

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

(b) During test of logic.

* When the system is required to be OPERABLE per Specification 3.5.1, 3.5.3.1 or 3.5.3.2.

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3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation trip systems shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the trip system inoperable until the trip system is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE trip systems less than required by the Minimum OPERABLE Trip Systems per Operating Pump requirement for one Trip Function, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1. Each ATWS recirculation pump trip system instrumentation trip system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

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TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE TRIP SYSTEMS PER OPERATING PUMP[(a)]
1. Reactor Vessel Water Level - Low Low, Level 2	1
2. Reactor Vessel Pressure - High	1

(a) One trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided the other trip system for that operating pump is OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	$\geq$ -(38) inches*	$\geq$ -( ) inches
2. Reactor Vessel Pressure - High	$\leq$ (1120) psig	$\leq$ ( ) psig

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TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	M	R
2. Reactor Vessel Pressure - High	NA	M	R

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END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as show in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is greater than or equal to (30)% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, operation may continue; restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

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SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2. LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of both trip systems shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months.

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TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Turbine Stop Valve - Closure	2[(b)]
2. Turbine Control Valve-Fast Closure	2[(b)]

(a)One trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

(b)These functions are bypassed when turbine first stage pressure is less than or equal to (   ) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALVE
1. Turbine Stop Valve-Closure	≤ (5)% closed	≤ (7)% closed
2. Turbine Control Valve-Fast Closure	≥ (500) psig	≥ (414) psig

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TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)
1. Turbine Stop Valve-Closure	$\leq$ ( )
2. Turbine Control Valve-Fast Closure	$\leq$ ( )

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	M*	R
2. Turbine Control Valve-Fast Closure	M*	Q

*Including trip system logic testing.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than (113) psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

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TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	2	50
b. Reactor Vessel Water Level - High	2[b]	50
c. Condensate Storage Tank Water Level - Low	(2)[b]	51
d. Manual Initiation	(2)	52

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b) One trip system with two-out-of-two logic.

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TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.

b. For both trip systems, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 24 hours or declare the RCIC system inoperable.

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TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -(38) inches*	$\geq$ -( ) inches*
b. Reactor Vessel Water Level - High	$\leq$ ( ) inches*	$\leq$ ( ) inches*
c. Condensate Storage Tank Level - Low	( ) inches	( ) inches
d. Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.

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TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R
b. Reactor Vessel Water Level - High	S	M	R
c. Condensate Storage Tank Level - Low	NA	M	Q
d. Manual Initiation	NA	M[(a)]	NA

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

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3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.6.1 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

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TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(g)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5**	62
b. Scram Trip Bypass	1	1, 2, 5**	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62

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TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

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TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ ((0.66) W + (41)%)	$\leq$ ((0.66) W + (43)%)
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5/125) of full scale	$\geq$ (3/125) of full scale
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	$\leq$ ((0.66) W + (42)%)*	$\leq$ ((0.66) W + (45)%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ (12)% of RATED THERMAL POWER	$\leq$ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (2 x 10^5) cps	$\leq$ (5 x 10^5) cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ (3) cps	$\geq$ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (108/125) of full scale	$\leq$ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ (5/125) of full scale	$\geq$ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	$\leq$ (18) gallons	$\leq$ (18) gallons
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	______	______
b. Inoperative	NA	NA
c. Comparator	______	______

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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TABLE 4.3.6.1-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b],M	Q	1*
b. Inoperative	NA	S/U[b],M	NA	1*
c. Downscale	NA	S/U[b],M	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U[b],M	Q	1
b. Inoperative	NA	S/U[b],M	NA	1, 2, 5
c. Downscale	NA	S/U[b],M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U[b],M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W[c]	NA	2, 5
b. Upscale	NA	S/U[b],W[c]	Q	2, 5
c. Inoperative	NA	S/U[b],W[c]	NA	2, 5
d. Downscale	NA	S/U[b],W[c]	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W[c]	NA	2, 5
b. Upscale	NA	S/U[b],W[c]	Q	2, 5
c. Inoperative	NA	S/U[b],W[c]	NA	2, 5
d. Downscale	NA	S/U[b],W[c]	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	1, 2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[b],M	Q	1
b. Inoperative	NA	S/U[b],M	NA	1
c. Comparator	NA	S/U[b],M	Q	1

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TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

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

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Radioactive Liquid Waste System Effluent Radiation Monitor	1	(a)	___	(10^{-1} to 10^{6}) cps	70
2. Service Water Effluent Radiation Monitor	1	At all times	___	(10^{-1} to 10^{6}) cps	71
3. Reactor Building Closed Cooling Water Radiation Monitor	1	At all times	___	(10^{-1} to 10^{6}) cps	71
4. Service Water Discharge, RHR Heat Exchanger, Radiation Monitor	1/heat exchanger	(b)	___	(10^{-1} to 10^{6}) cps	71
5. Reactor Building Vent Radiation Monitor	$3^{(c)}$	At all times	___$^{(d)}$	(10^{-2} to 10^{2}) mR/hr	72
6. Plant Vent Stack Plenum Exhaust Radiation Monitor	1	At all times	___	(0.01 to 100) mR/hr	71
7. Main Condenser Air Ejector Off-Gas Post Treatment Radiation Monitor	$2^{(c)}$	1, 2	___$^{(e)}$	(1 to 10^{6}) cps	73
8. Main Control Room Ventilation Radiation Monitor	2/intake	1,2,3,5 and *	(5) mR/hr	(0.1 to 10,000) mR/hr	74

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TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
9. Fuel Pool Ventilation Radiation Monitor	$3^{(c)}$	(f)	$\leq$ (2) x background (g)	$(10^{-2}$ to $10^{2})$ mR/hr	75
10. Standby Gas Treatment System					
a. Reactor Bldg. Vent Radiation Monitor	$3^{(c)}$	1, 2, 3, 5 and *	____	$(10^{-2}$ to $10^{2})$ mR/hr	76
b. Fuel Pool Ventilation Radiation Monitor	$3^{(c)}$	(f)	$\leq$ (2) x background	$(10^{-2}$ to $10^{2})$ mR/hr	76
11. Area Monitors					
a. Criticality Monitors					
1) New Fuel Storage Vault	1	(h)	$\leq$ (15) mR/hr	$(10^{-1}$ to $10^{3})$ mR/hr	77
2) Spent Fuel Storage Pool	1	(i)	$\leq$ (15) mR/hr	$(10^{-1}$ to $10^{6})$ mR/hr	77
b. Main Control Room Radiation Monitor	1	At all times	____	$(10^{-2}$ to $10^{2})$ mR/hr	77

* When irradiated fuel or a spent fuel shipping cask is being handled in the secondary containment.
(a) With radioactive waste stored in or being discharged from the radioactive liquid waste storage system.
(b) With RHR heat exchangers in operation.
(c) Trips system with 2 channels upscale, or one channel upscale and one channel downscale, or 2 channels downscale.
(d) Also isolates the primary and secondary containment purge and vent penetrations, valve group(s) ( ).
(e) Time delay before valve closure ____ $\pm$ ____ seconds.
(f) With irradiated fuel in the spent fuel storage pool or building.
(g) Also isolates the secondary containment purge and vent penetrations, valve group(s) ( ).
(h) With fuel in the new fuel storage vault.
(i) With fuel in the spent fuel storage pool.

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TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 70 - With the required monitor inoperable, take and analyze two independent samples of each radwaste discharge tank to be discharged prior to release. Restore the inoperable monitor to OPERABLE status within 72 hours or suspend release of liquid radwaste.

ACTION 71 - With the required monitor inoperable, obtain and analyze at least one grab sample of the monitored parameter at least once per 24 hours.

ACTION 72 -

a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within one hour.

b. With two of the required monitors inoperable, shutdown the (primary and secondary)) containment ventilation systems and isolate the primary and secondary purge and vent penetrations within 12 hours.

ACTION 73 -

a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within one hour.

b. With both of the required monitors inoperable, be in at least HOT SHUTDOWN within 12 hours.

ACTION 74 -

a. With one of the required monitors inoperable, place the inoperable channel in the (downscale) tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the (pressurization) mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the (pressurization) mode of operation within one hour.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION (Continued)

ACTION 75 -

a. With one of the required monitors inoperable, place the inoperable channel in the (downscale) tripped condition within one hour.

b. With two of the required monitors inoperable, shutdown the fuel pool ventilation system and isolate the secondary containment purge and vent penetrations within 12 hours.

ACTION 76 -

a. With one of the required monitors inoperable, place the inoperable channel in the (downscale) tripped condition within one hour.

b. With two of the required monitors inoperable, initiate and maintain operation of at least one standby gas treatment subsystem within 12 hours.

ACTION 77 With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

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TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION#	CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Radioactive Waste System Effluent Radiation Monitor	S	M	R	(a)
2. Service Water Effluent Radiation Monitor	S	M	R	At all times
3. Reactor Building Closed Cycle Water Radiation Monitor	S	M	R	At all times
4. Service Water Discharge, RHR Heat Exchanger, Radiation Monitor	S	M	R	(b)
5. Reactor Building Vent Radiation Monitor	S	M	R	At all times
6. Plant Vent Stack Plenum Exhaust Radiation Monitor	S	M	R	At all times
7. Main Condenser Air Ejector Off-Gas Post Treatment Radiation Monitor	S	M	R	1, 2
8. Main Control Room Ventilation Radiation Monitor	S	M	R	1, 2, 3, 5 and *

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION#	CONDITIONS IN WHICH SURVEILLANCE REQUIRED
9. Fuel Pool Ventilation Radiation Monitor	S	M	R	(c)
10. Standby Gas Treatment System				
a. Reactor Bldg. Vent Radiation Monitor	S	M	R	1, 2, 3, 5 and *
b. Fuel Pool Ventilation Radiation Monitor	S	M	R	(c)
11. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	S	M	R	(d)
2) Spent Fuel Storage Pool	S	M	R	(e)
b. Main Control Room Radiation Monitor	S	M	R	At all times

(a) With radioactive waste stored in or being discharged from the radioactive liquid waste storage system.
(b) With RHR heat exchangers in operation.
(c) With irradiated fuel in the spent fuel storage pool or building.
(d) With fuel in the new fuel storage vault.
(e) With fuel in the spent fuel storage vault.
* When irradiated fuel or a spent fuel shipping cask is being handled in the secondary containment.
\# The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

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SEISMIC MONITORING INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

3.3.7.2 The seismic monitoring instrumentation shown in Table 3.3.7.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more seismic monitoring instruments inoperable for more than 30 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.7.2.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.2-1.

4.3.7.2.2 Each of the above required seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. In lieu of any other report required by Specification 6.9.1, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety.

*This specification not required for additional units at a common site provided at least one unit has seismic instrumentation and corresponding technical specifications meeting the recommendations of Regulatory Guide 1.12, April 1974.

TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accele:ographs		
a.		1
b.		1
c.		1
d.		1
2. Triaxial Peak Accelerographs		
a.		1
b.		1
c.		1
d.		1
3. Triaxial Seismic Switches		
a.		1(a)
b.		1(a)
c.		1(a)
4. Triaxial Response-Spectrum Recorders		
a.		1(a)
b.		1

(a) With reactor control room indication and annunciation.

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TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs			
a. ____________	M	SA	R
b. ____________	M	SA	R
c. ____________	M	SA	R
d. ____________	M	SA	R
2. Triaxial Peak Accelerographs			
a. ____________	NA	NA	R
b. ____________	NA	NA	R
c. ____________	NA	NA	R
d. ____________	NA	NA	R
3. Triaxial Seismic Switches			
a. ____________	M[(a)]	SA	R
b. ____________	M	SA	R
c. ____________	M	SA	R
4. Triaxial Response-Spectrum Recorders			
a. ____________	M	SA	R
b. ____________	NA	SA	R

(a) Except seismic trigger.

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METEOROLOGICAL MONITORING INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

3.3.7.3 The meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more meteorological monitoring instrumentation channels inoperable for more than 7 days, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.7.3 Each of the above required meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.3-1.

*This specification not required for additional units at a common site provided at least one unit has meteorological instrumentation and the corresponding technical specifications and that the same meteorological data is applicable to the additional units.

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
1. Wind Speed	
1. Elev. (30) ft.	1
2. Elev. (200) ft.	1
2. Wind Direction	
1. Elev. (30) ft.	1
2. Elev. (200) ft.	1
3. Air Temperature	
1. Elev. (30) ft.	1
2. Elev. (200) ft.	1
4. Air Temperature Difference	
1. Elev. (30/200) ft.	1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
1. Elev. (30) ft.	D	SA
2. Elev. (200) ft.	D	SA
2. Wind Direction		
1. Elev. (30) ft.	D	SA
2. Elev. (200) ft.	D	SA
3. Air Temperature		
1. Elev. (30) ft.	D	SA
2. Elev. (200) ft.	D	SA
4. Air Temperature Difference		
1. Elev. (30/200) ft.	D	SA

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown monitoring instrumentation channels shown in Table 3.3.7.4-1 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more remote shutdown monitoring instrumentation channels inoperable, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

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TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	________	1
2. Reactor Vessel Water Level	________	1
3. Safety/Relief Valve Position, (2) valves	________	1(/valve)
4. Suppression Chamber Water Level	________	1
5. Suppression Chamber Water Temperature	________	1
6. Suppression Chamber Air Temperature	________	1
7. Drywell Pressure	________	1
8. Drywell Temperature	________	1
9. RHR System Flow	________	1
10. RHR Service Water System Flow	________	1
11. RHR Service Water Temperature	________	1
12. RCIC System Flow	________	1
13. RCIC Turbine Speed	________	1
14. ____________________	________	–

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TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Safety/Relief Valve Position	M	NA
4. Suppression Chamber Water Level	M	R
5. Suppression Chamber Water Temperature	M	R
6. Suppression Chamber Air Temperature	M	R
7. Primary Containment Pressure	M	R
8. Drywell Temperature	M	R
9. RHR System Flow	M	R
10. RHR Service Water System Flow	M	R
11. RHR Service Water Temperature	M	R
12. RCIC System Flow	M	R
13. RCIC Turbine Speed	M	R
14. ______	______	______

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POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The post accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more post-accident monitoring instrumentation channels inoperable, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required post-accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.5-1.

TABLE 3.3.7.5-1

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	2
2. Reactor Vessel Water Level	2
3. Suppression Chamber Water Level	2
4. Suppression Chamber Water Temperature	2
5. Suppression Chamber Air Temperature	2
6. Drywell Pressure	2
7. Drywell Temperature	2
8. Drywell Oxygen Concentration	2
9. Drywell Hydrogen Concentration	2
10. ______________	______
11. ______________	______
12. ______________	______

TABLE 4.3.7.5-1

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure	M	R
2. Reactor Vessel Water Level	M	R
3. Suppression Chamber Water Level	M	R
4. Suppression Chamber Water Temperature	M	R
5. Suppression Chamber Air Temperature	M	R
6. Primary Containment Pressure	M	R
7. Drywell Temperature	M	R
8. Drywell Oxygen Concentration	NA	R
9. Drywell Hydrogen Concentration	NA	R
10.		
11.		
12.		

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POOR ORIGINAL

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least three source range monitor channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERABLE CONDITION 3 or 4 with two or more of the above required source range monitor channels inoperable, verify all control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

 1. CHANNEL CHECK at least once per:

  a) 12 hours in CONDITION 2*, and

  b) 24 hours in CONDITION 3 or 4.

 2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

 1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

 2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3 cps with the detector fully inserted.

*With IRM's on range 2 or below.
**Neutron detectors may be excluded from CHANNEL CALIBRATION.

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TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe system shall be OPERABLE with:

a. Three movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all three detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors,

b. Monitoring the APLHGR, LHGR, or MCPR, and

c. Adjustment of the APRM setpoints.

ACTION:

With the traversing in-core probe system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to or during use when required for the above applicable monitoring or calibration functions.

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 Two independent chlorine detection system subsystems, each with two detectors, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to (5) ppm, shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one chlorine detector in one or both chlorine detection subsystems inoperable, restore the inoperable detector(s) to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the (pressurization) mode of operation.

b. With one or more chlorine detection subsystems inoperable, within one hour initiate and maintain operation of the control room emergency filtration system in the (pressurization) mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each of the above required chlorine detection system subsystems shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

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CHLORIDE INTRUSION MONITORS (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.3.7.9 The chloride intrusion monitor channels shown in Table 3.3.7.9-1 shall be OPERABLE with alarm setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.7.9-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION

a. With a chloride intrusion monitor channel trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.3.7.9-2, declare the monitor inoperable until the monitor is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than the Minimum OPERABLE Channels for up to two functional units, sample the parameter monitored by the inoperable channel(s) of the functional unit(s) at least once per 4 hours; restore at least the Minimum OPERABLE Channels for at least two functional units to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

c. With the number of OPERABLE Channels less than the Minimum OPERABLE Channels for more than two functional units, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.7.9 Each of the above required chloride intrusion monitors shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.9-1.

1069 256

TABLE 3.3.7.9-1

CHLORIDE INTRUSION MONITORS

FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS
1. Chloride detectors in the condenser hotwell outlet headers	(4)
2. Chloride detectors in the condensate pump discharge	1
3. Chloride detector in the inlet to the deep bed demineralizer	1

1069 257

TABLE 3.3.7.9-2

CHLORIDE INTRUSION MONITORS SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. Chloride detectors in the condenser hotwell outlet headers	≤ (1.0) μmhos/cm
2. Chloride detectors in the condensate pump discharge	≤ (0.3) μmhos/cm (2.0 μmhos/cm for wide range monitor)
4. Chloride detector in the inlet to the deep bed demineralizer	≤ (0.3) μmhos/cm

1069 258

TABLE 4.3.7.9-1

CHLORIDE INTRUSION MONITORS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Chloride detectors in the condenser hotwell outlet headers	D	M	R
2. Chloride detectors in the condensate pump discharge	D	M	R
3. Chloride detector in the inlet to the deep bed demineralizer	D	M	R

1069 259

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.10 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.10-1 shall be OPERABLE.

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

ACTION:

With the number of OPERABLE fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.10-1:

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the primary containment at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.7).

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

4.3.7.10.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.7.10.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

1069 260

TABLE 3.3.7.10-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION (Illustrative)**	MINIMUM INSTRUMENTS OPERABLE* HEAT	FLAME	SMOKE
a. Containment			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
b. Control Room			
c. Cable Spreading			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
d. Computer Room			
e. Switchgear Room			
f. Remote Shutdown Panels			
g Station Battery Rooms			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
h. Turbine			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
i. Diesel Generator			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
j. Diesel Fuel Storage			
k. Safety Related Pumps			
Zone 1 Elevation ____			
Zone 2 Elevation ____			
l. Fuel Storage			
Zone 1 Elevation ____			
Zone 2 Elevation ____			

*The fire detection instruments located within the primary containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

**List all detectors in areas required to insure the OPERABILITY of safety related equipment and indicate instruments which automatically actuate fire suppression system.

1069 261

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

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

a. At least once per 7 days by:

   1. Cycling each of the following valves through at least one complete cycle from the running position:

      a) For the overspeed protection control system;

         1) Four high pressure turbine control valves, and

         2) Four low pressure turbine interceptor valves

      b) For the electrical overspeed trip system and the mechanical overspeed trip system:

         1) Four high pressure turbine throttle stop valves,

         2) Four high pressure turbine reheat stop valves,

         3) Four high pressure turbine control valves, and

         4) Four low pressure turbine interceptor valves.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

1069 263

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure, and

b. Verifying that the rate of control valve movement is:

1. Less than or equal to ( )% of stroke per second opening, and

2. Less than or equal to ( )% of stroke per second closing.

*See Special Test Exception 3.10.4.

1069 264

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by verifying that the following conditions do not occur (simultaneously) when the recirculation pumps are operating (at the same speed):

a. The recirculation pump flow differs by more than (10)% from the established valve position-loop flow characteristics.

b. The indicated total core flow differs by more than (10)% from the established total core flow value derived from recirculation loop flow measurements.

c. The diffuser-to-lower plenum differential pressure reading on any individual jet pump differs from the mean of all jet pump differential pressures, in the same loop, by more than (10)%.

1069 265

RECIRCULATION PUMPS (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speeds shall be maintained within:

a. (10)% of each other with THERMAL POWER great than or equal to (80)% of RATED THERMAL POWER.

b. (15)% of each other with THERMAL POWER less than (80)% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

a. Restore the pump speeds to within the specified limit within (30) minutes, or

b. Declare the recirculation loop of the pump with the slower speed not in operation and take the ACTION require by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation pump speeds shall be verified to be within the limits at least once per 24 hours.

1069 266

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor coolant within the dome and the bottom head drain is less than or equal to (100)°F, and:

a. The temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to (50)°F when both loops have been idle, or

b. The temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to (50)°F when only one loop has been idle, and the operating loop flow rate is less than or equal to (50)% of rated loop flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 30 minutes prior to startup of an idle recirculation loop.

1069 267

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 (At least (two) reactor coolant system code safety valves and) the code safety valve function of (at least) (11) reactor coolant system safety-relief valves shall be OPERABLE with the following lift settings.*

((2) safety valves @ (1130) psig ±1%)
(3) safety-relief valves @ (1175) psig ±1%
(3) safety-relief valves @ (1185) psig ±1%
(3) safety-relief valves @ (1195) psig ±1%
(2) safety-relief valves @ (1205) psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With (one or more of the above required reactor coolant system code safety valves or with) the code safety valve function of one or more of the above required safety-relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2 The code safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE by verifying that the bellows on the safety/relief valves have integrity, by instrumentation indication, at least once per 24 hours.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

1069 268

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere (gaseous or particulate) radioactivity monitoring system,

b. The primary containment sump flow monitoring system, and

c. Either the (primary containment air coolers condensate flow rate monitoring system) or the primary containment atmosphere (gaseous or particulate) radioactivity monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere particulate and/or gaseous monitoring system-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Primary containment air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 5 gpm UNIDENTIFIED LEAKAGE, and

c. 25 gpm total leakage averaged over any 24 hour period.

(d. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4 hour period.)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(c. With any reactor coolant system leakage greater than the limit in d above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.)

SURVEILLANCE REQUIREMENTS

4.4.3.2 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric particulate (and/or gaseous) radioactivity at least once per 12 hours,

b. Monitoring the primary containment sump flow rate at least once per 12 hours, and

c. Monitoring the primary containment air coolers condensate flow rate at least once per 12 hours.

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

1069 270

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With the conductivity or chloride concentration exceeding the limit specified in Table 3.4.4-1, but less than 10 μmho/cm at 25°C and less than 0.5 ppm, respectively, operation may continue for up to 72 hours and this need not be reported to the Commission provided that operation under these conditions shall not exceed 336 hours per year The provisions of Specification 3.0.4 are not applicable.

2. With the conductivity, chloride concentration or PH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity exceeding 10 μmho/cm at 25°C or chloride exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. At all other times:

1. With the conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours.

2. With the chloride concentration limit of Table 3.4.4-1 exceeded for more than 24 hours, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 3.

3. The provisions of Specification 3.0.3 are not applicable.

1069 271

SURVEILLANCE REQUIREMENTS

4.4.4 The reactor coolant shall be determined to be within the specified chemistry limit by:

a. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 hours.

b. Analyzing a sample of the reactor coolant:

   1. Chlorides at least once per:

      a) 72 hours, and

      b) 8 hours whenever conductivity is greater than 1.0 μmho/cm at 25°C.

   2. Conductivity at least once per 72 hours.

   3. pH at least once per:

      a) 72 hours, and

      b) 8 hours whenever conductivity is greater than 1.0 μmho/cm at 25°C.

c. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, obtaining an in-line conductivity measurement at least once per:

   1. 4 hours in OPERATIONAL CONDITIONS 1, 2 and 3, and

   2. 24 hours at all other times.

d. Performance of a CHANNEL CALIBRATION of the continuous conductivity monitor with an in-line flow cell at least once per:

   1. 7 days, and

   2. 24 hours whenever conductivity is greater than 1.0 μmho/cm at 25°C.

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TABLE 3.4.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

OPERATIONAL CONDITION	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)	PH
1	$\leq$ 0.2 ppm	$\leq$ 1.0	5.6 $\leq$ pH $\leq$ 8.6
2 and 3	$\leq$ 0.1 ppm	$\leq$ 2.0	5.6 $\leq$ pH $\leq$ 8.6
At all other times	$\leq$ 0.5 ppm	$\leq$ 10.0	5.3 $\leq$ pH $\leq$ 8.6

POOR ORIGINAL

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. Greater than 100/$\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

b. In OPERABLE CONDITIONS 1, 2, 3 or 4;

1. With the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. With:

   a) THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour, or

   b) The off-gas level, at the SJAE, increased by more than (10,000) microcuries per second in one hour at release rates less than (75,000) microcuries per second, or

   c) The off-gas level, at the SJAE, increased by more than (15)% in one hour at release rates greater than (75,000) microcuries per second, or

   perform the sampling and analysis requirements of Item 4C of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.2 at least once per 92 days containing the results of the specific activity analysis together with the below additional information for each occurrence.

Additional Information

1. Reactor power history starting 48 hours prior to:

   a) The first sample in which the limit was exceeded, and/or

   b) The THERMAL POWER or off-gas level change.

2. Fuel burnup by core region.

3. Clean-up flow history starting 48 hours prior to:

   a) The first sample in which the limit was exceeded, and/or

   b) The THERMAL POWER or off-gas level change.

4. Off-gas level starting 48 hours prior to:

   a) The first sample in which the limit was exceeded, and/or

   b) The THERMAL POWER or off-gas level change.

SURVEILLANCE REQUIREMENTS

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

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TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133 and I-135	a) At least once per 31 days	1
	b) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.1.	1#, 2#, 3#, 4#
	c) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION b.2.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

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

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

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3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curves A and A' for hydrostatic or leak testing; (2) curves B and B' for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C and C' for operations with a critical core other than low power PHYSICS TESTS, with:

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

b. A maximum cooldown of (100)°F in any one hour period, and

c. The reactor vessel flange and head flange temperature greater than or equal to (70)°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations:

a. The reactor coolant system temperature and pressure shall be determined to be within the heatup and cooldown limits and to the right of the limits of Figure 3.4.6.1-1 curves A and A' or B and B', as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

(b. The reactor coolant system temperature at the following locations shall be recorded at least once per 5 minutes until 3 successive readings at each location are within (5)°F:

1. Reactor vessel shell adjacent to shell flange,

2. Reactor vessel bottom drain,

3. Recirculation loops A and B, and

4. Reactor vessel bottom head.)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor flux wire specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER at the intervals required by 10 CFR 50, Appendix H. The results of these fluence determinations, in conjunction with Bases Figure B 3/4.4.6-2, shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to (70)°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

   2. $\leq$ (80)°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

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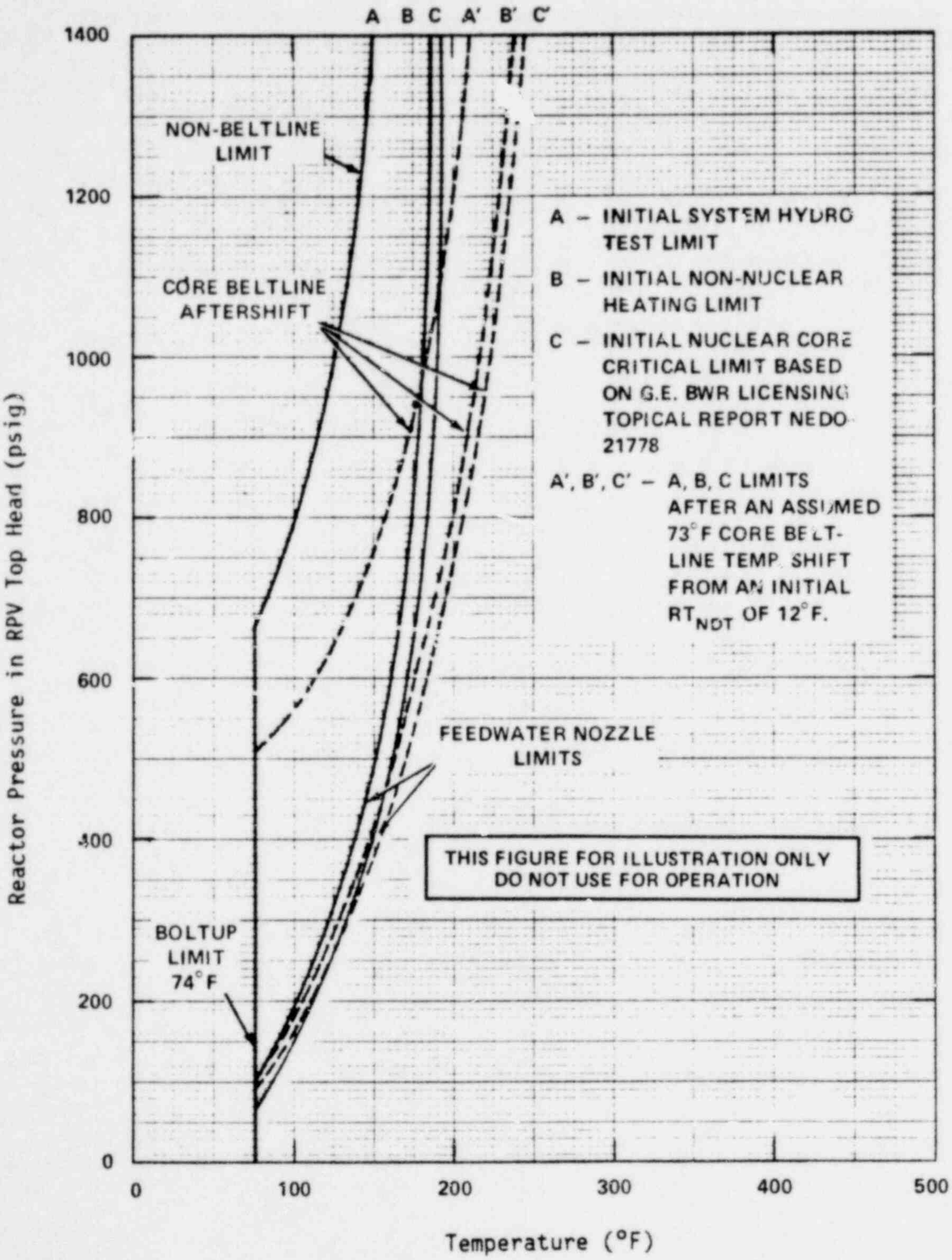
POOR ORIGINAL



MINIMUM TEMPERATURE VS. REACTOR VESSEL PRESSURE
Figure 3.4.6.1-1

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REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than (1045) psig.

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding (1045) psig, reduce the pressure to less than (1045) psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than (1045) psig at least once per 12 hours.

*Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to (3) and less than or equal to (5) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one or more MSIVs inoperable:

a. Operation may continue and the provisions of Specification 3.0.4 are not applicable provided that at least one MSIV is maintained OPERABLE in each affected main steam line that is open and within 8 hours, either:

   1. The inoperable valve(s) is restored to OPERABLE status or

   2. The affected main steam line is isolated by use of a deactivated MSIV in the closed position.

b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between (3) and (5) seconds when tested pursuant to Specification 4.0.5.

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3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 212°F.

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

d. The provisions of Specification 3.0.4 are not applicable.

e. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated per the requirements of Specification 4.0.5.

1069 282

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 and the automatic depressurization system (ADS) shall be OPERABLE with:

a. ECCS division 1 consisting of the:

1. OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

b. ECCS division 2 consisting of two OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

d. The automatic depressurization system having at least (seven) OPERABLE ADS valves.

APPLICABILITY: OPERATIONAL CONDITION 1, 2* # and 3*.

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to (113) psig.
#See Special Test Exception 3.10.5.

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LIMITING CONDITION FOR OPERATION (continued)

ACTION:

a. For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:

   1) With the LPCS system inoperable, POWER OPERATION may continue; restore the inoperable LPCS system to OPERABLE status within (15) days.

   2) With LPCI subsystem "A" inoperable, POWER OPERATION may continue; restore the inoperable LPCI subsystem to OPERABLE status within (41) days.

   3) With ECCS division 1 inoperable, POWER OPERATION may continue: restore the inoperable division to OPERABLE status with (72) hours.

   4) Or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:

   1) With LPCI subsystem "B" or "C" inoperable, POWER OPERATION may continue; restore the inoperable LPCI subsystem "B" and "C" to OPERABLE status within (41) days.

   2) With ECCS division 2 inoperable, POWER OPERATION may continue, restore the inoperable division to OPERABLE status within (72) hours.

   3) Or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

c. For ECCS division 3, provided that ECCS divisions 1 and 2, the ADS and the RCIC system are OPERABLE:

   1) With ECCS division 3 inoperable, POWER OPERATION may continue; restore the inoperable division to OPERABLE status within (11) days

   2) Or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN within the time limit of this ACTION or as applicable, maintain reactor coolant temperature less than or equal to (400)°F by use of alternate heat removal methods.

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LIMTING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:

1) With LPCI subsystem "A" and LPCI subsystem "B" and "C" inoperable, POWER OPERATION may continue; restore at least the inoperable LPCI subsystem "A" or the inoperable LPCI subsystem "B" and "C" to OPERABLE status within (7) days.

2) With the LPCS system inoperable and LPCI subsystems "B" and "C" inoperable, POWER OPERATION may continue; restore at least the inoperable LPCS system or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3) Or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

e. With one of the above required ADS valves inoperable, POWER OPERATION may continue; restore the inoperable ADS valve to OPERABLE status within (14) days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ (113) psig within the following 24 hours.

f. With an (ECCS) discharge line "keep filled" pressure instrumentaton channel inoperable, perform Surveillance Requirement 4.5.1.a.1 at least once per 24 hours.

g. With an ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status with 72 hours or declare the associated ECCS inoperable.

h. With the Surveillance Requirement of Specification (4.5.1.d.2) not performed at the required internal due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the appropriate surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

i. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

*Whenever two or mone RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN within the time limit of this ACTION or as applicable, maintain reactor coolant temperature less than or equal to (400)°F by use of alternate heat removal methods.

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SURVEILLANCE REQUIREMENTS

4.5.1 ECCS division 1, 2 and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI and HPCS systems:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Performance of a CHANNEL FUNCTIONAL TEST of the:

a) Discharge line "keep filled" pressure instrumentation, and

b) Header delta P instrumentation.

3. Verifing that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secure in position, is in its correct position.

b. Verifing that, when tested pursuant to specification 4.0.5, each:

1. LPCS pump develops a flow of at least (6598) gpm against a discharge pressure greater than or equal to (452) psig.

2. LPCI pump develops a total flow of at least (7666) gpm against a discharge pressure greater than or equal to (111) psig.

3. HPCS pump develops a flow of at least (659) gpm against a discharge head of greater than or equal to (397) psig.

c. At least once per 18 months for the LPCS, LPCI and HPCS systems:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Performing a CHANNEL CALIBRATION of the:

a) Discharge line "keep filled" pressure instrumentation and verifying the:

1) High pressure setpoint and the low pressure setpoint of the:

(a) LPCS system to be $\leq$ (450) psig and $\geq$ (40) psig, respectively.

(b) LPCI system to be $\leq$ (400) psig and $\geq$ (40) psig, respectively.

2) Low pressure setpoint of the HPCS system to be $\geq$ (40) psig.

b) Header delta P instrumentation and verifying the setpoint of the:

1) LPCS system and LPCI subsystem (A) to be $\pm$ (1) psid.

2) LPCI subsystem (B) and (C) to be $\pm$ (1) psid.

3) HPCS system to be (5) $\pm$ (1.5) psid greater than the normal indicated ΔP.

3. Verifying that the suction for the HPCS system is (automatically) transferred from the condensate storage tank to the suppression chamber on a condensate storage tank low water level signal and on a suppression chamber high water level signal.

d. At least once per 18 months for the ADS by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

2. Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig and observing that either:

a) The control valve or bypass valve position responds accordingly, or

b) There is a corresponding change in the measured steam flow.

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3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least one of the following a, b or c shall be OPERABLE:

a. ECCS division 1 consisting of the:

1. OPERABLE low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

b. ECCS division 2 consisting of two OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.

c. Both of the following:

1. ECCS division 3 consisting of the high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

   a) From the suppression chamber, or

   b) When the suppression pool is drained, from the condensate storage tank containing at least (150,000) gallons of water.

2. At least one pump and flow path from ECCS division 1 and/or 2.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

*ECCS divisions 1, 2 and 3 and the suppression chamber are not required to be OPERABLE provided tht the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For ECCS division 1, with ECCS divisions 2 and 3 inoperable:

1. With the LPCS system inoperable or LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" or the inoperable, LPCS system to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

2. With the LPCS system inoperable and LPCI subsystem "A" inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least the HPCS or LPCS system or one LPCI subsystem to OPERABLE status within 4 hours or, in OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

b. For ECCS division 2 with ECCS divisions 1 and 3 inoperable:

1. With LPCI subsystem "B" and "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

2. With the LPCI subsystems "B" and "C" inoperable, suspend CORE ALTERNATIONS and all operations that have a potential for draining the reactor vessel. Restore at least the HPCS or LPCS systems or one LPCI subsystem to OPERABLE status within 4 hours or, in OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

c. For ECCS division 3 with ECCS division 1 and 2 inoperable:

1. With the HPCS system OPERABLE, restore at least the LPCS system or one LPCI subsystem to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

2. With the HPCS system inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least the HPCS or LPCS system or one LPCI subsystem to OPERABLE status within 4 hours, or, in OPERATIONAL CONDITION 4, establsih SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 At least the above required ECCS divisions shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

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3/4.5.3 SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE with minimum contained water volume of (142,160) ft3, equivalent to a level of (26'10") and the water level instrumentation channel alarm adjusted to actuate at a low water level greater than or equal to ( ), except that the suppression chamber may be drained in OPERATIONAL CONDITION 4 and 5* provided that:

a. No operations are performed that have a potential for draining the reactor vessel,

b. The reactor mode switch is locked in the Shutdown or Refuel position,

c. The condensate storage tank contains at least (150,000) gallons of water, and

d. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 and 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. In OPERATIONAL CONDITION 4 establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. The provisions of Specification 3.0.3 are not applicable.

*ECCS divisions 1, 2 and 3 and the suppression chamber are not required to be OPERABLE provided that the reactor vessel head is removed and the cavity is flooded, the spent fuel pool gates are removed, and the water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 30 days or verify the suppression chamber water level to be greater than or equal to (26'10") at least once per 12 hours by an alternate method.

d. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression chamber water level to be greater than or equal to (26'10") at least once per hour by at least one alternate method.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to (26'10") at least once per 12 hours.

b. Two suppression chamber water level instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months.

4.5.3.2 With the suppression chamber drained, at least once per 12 hours:

a. Verify the above required conditions to be satisfied, or

b. Verify the footnote conditions to be satisfied.

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals with gas at Pa, (40.4) psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

b. At least once per 31 days by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3.1-1 of Specification 3.6.3.

c. By verifying each containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

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*See Special Test Exception 3.10.1

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

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PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, (0.635) percent by weight of the containment air per 24 hours at P_a, (40.4) psig, or

2. Less than or equal to L_t, (0.90) percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (20.2) psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests when pressurized to P_a.

c. *Less than or equal to (11.5) scf per hour for any one main steam isolation valve when tested at P_t, (20.2) psig.

d. *Less than or equal to (   ) scf per hour for any one feedwater isolation valve when tested at (   ) psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or

b. the measured combined leakage rate for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests exceeding 0.60 L_a, or

c. the measured leak rate exceeding (11.5) scf per hour for any one main steam isolation valve, or

d. the measured leak rate exceeding (   ) scf per hour for any one feedwater isolation valve,

restore:

a. the overall integrated leakage rate(s) to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and

*Exemption to Appendix "J" of 10 CFR 50.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. the combined leakage rate for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. the leakage rate to less than or equal to (11.5) scf per hour for any one main steam isolation valve, and

d. the leakage rate to less than or equal to ( ) scf per hour for any one feedwater isolation valve,

prior to increasing reactor coolant system temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - (1972):

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at either P_a, (40.4) psig, or at P_t, (20.2) psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a or 0.25 L_t.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Has duration sufficient to establish accurately the change in leakage rate between the type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, (40.4) psig, or P_t, (20.2) psig.

d. Type B and C tests shall be conducted with gas at P_a, (40.4) psig, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves and feedwater isolation valves,

3. Penetrations using continuous leakage monitoring systems, and

4. Valves pressurized with fluid from a seal system.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line and feedwater isolation valves shall be leak tested at least once per 18 months.

g. Type B periodic tests are not required for penetrations continuously monitored by the Containment Penetration Pressurization System, provided the system is OPERABLE per Specification 3.6.1.4.

h. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, (44.4) psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

i. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, (40.4) psig, at intervals no greater than once per 3 years.

j. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurements system.

k. The provisions of Specification 4.0.2 are not applicable.

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PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, (40.4) psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed; operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days. The provisions of Specification 3.0.4 are not applicable.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*See Special Test Exception 3.10.1.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. **After opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to (5) scf per hour when the gap between the door seals is pressurized to Pa, (40.4) psig.

b. At least once per 6 months# by conducting an overall air lock leakage test at P_a, (40.4) psig and by verifying that the overall air lock leakage rate is within its limit.

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

**Exemption to Appendix "J" of 10 CFR 50.

The provisions of Specification 4.0.2 are not applicable.

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MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two MSIV leakage control svstem (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting the blower from the control room and operating the blower for at least 15 minutes.

2. Cycling each (air dilution valve) through at least one complete cycle of fuel travel.

3. (Heater OPERABILITY by) ______________________________.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each bleeder valve and steam isolation valve through at least one complete cycle of full travel.

c. At least once per 18 months by performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and the blower starts and develops at least the below required vacuum at the rated capacity:

1. Inboard valves - (60)" H_2O at (100) scfm.

2. Outboard valves - (50)" H_2O at (240) scfm.

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PRIMARY CONTAINMENT STRUCTURAL INTEGRITY (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

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POOR ORIGINAL

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY (Prestressed Concrete Containment with Ungrouted Tendons)

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 Primary Containment Tendons: The structural integrity of the primary containment tendons shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter, by:

a. Determining that a representative sample* of at least (15) tendons, (5) vertical and (10) hoop, each have a lift off force of between ____ (minimum) and ____ (maximum) pounds at the first year inspection. For subsequent inspections, the maximum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount _______ log t and the minimum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount _____ log t, where t is the time interval in years from initial tensioning of the tendon to the current testing date. This test shall include an unloading cycle in which each of these tendons is detensioned to determined if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to their observed lift off force, ± 3%. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously. If the lift off force of any one tendon in the total sample population is out of the predicted bounds, less than minimum or greater than maximum, an adjacent tendon on each side of the defective tendon shall also be checked for lift off force.

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group, vertical and hoop, may be kept unchanged after the initial selection.

POOR ORIGINAL

SURVEILLANCE REQUIREMENTS (Continued)

If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. This single tendon shall be restored to the required level of integrity. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the containment structure. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons checked for lift off force and change in elongation during subsequent tests may be reduced to a representative sample of at least 9 tendons, 3 vertical and 3 hoop.

b. Removing one wire or strand from each vertical and hoop tendon checked for a lift off force and determining over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage.

2. There are no changes in physical appearance of the sheathing filler grease.

3. A minimum tensile strength value of ____ psi guaranteed ultimate strength of the tendon material for at least three wire or strand samples, one from each end and one at mid-length, cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensil strength test is evidence of abnormal degradation of the containment structure.

4.6.1.5.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.5.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests while the containment is at its maximum test pressure.

4.6.1.5.3 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of these surfaces. The inspection shall be performed prior to the Type A Containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.4 Reports Any abnormal degradation of the containment structure detected during the above require tests and inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the tendon condition, the condition of the concrete, especially at tendon anchorages, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

1069 301

PRIMARY CONTAINMENT IN ERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Primary containment internal pressure shall be maintained between -(2.0) and +(2.0) psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the primary containment internal pressure outs de of the specified limits, restore the internal pressure to within the limit w thin 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

1069 302

PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

POOR ORIGINAL

LIMITING CONDITION FOR OPERATION

3.6.1.7 Primary containment average air temperature shall not exceed (110)°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the primary containment average air temperature greater than (110)°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The primary containment average air temperature shall be the (arithmetrical) average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	________	________
b.	________	________
c.	________	________
d.	________	________
e.	________	________
f.	________	________

1069 303

3/4.6.2 DEPRESSURIZATION SYSTEMS

POOR ORIGINAL

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with the pool water:

a. Volume between (87,600) ft^3 and (89,600) ft^3, equivalent to a level between (22' 0") and (22'6"), and a

b. Maximum temperature of (95)°F during OPERATIONAL CONDITION 1 or 2, except that the maximum temperature may be permitted to increase to:

1. (105)°F during testing which adds heat to the suppression chamber during OPERATIONAL CONDITION 1 or 2.
2. (110)°F with THERMAL POWER less than or equal to (1)% of RATED THERMAL POWER.
3. (120)°F with the main steam line isolation valves closed following a scram from OPERATIONAL CONDITION 1 or 2.

c. Level instrumentation channel alarms adjusted to actuate at a:

1. High water level less than or equal to ( ), and
2. Low water level greater than or equal to ( ).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber water temperature greater than (95)°F, except as permitted above, restore the temperature to less than or equal to (95)°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. In OPERATIONAL CONDITION 1 or 2 with the suppression chamber water temperature greater than (105)°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the temperature to less than (95)°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. In OPERATION CONDITION 1 or 2 with the suppression chamber water temperature greater than (110)°F and THERMAL POWER less than (1)% of RATED THERMAL POWER, place the reactor mode switch in the Shutdown position.

e. With the suppression chamber water temperature greater than (120)°F and with the main steam line isolation valves closed following a scram from OPERATIONAL CONDITION 1 or 2, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

f. With one suppression chamber water level instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 30 days or verify suppression chamber water level to be within the limits by an alternate method at least once per 12 hours.

g. With both suppression chamber water level instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber water temperature to be less than or equal to (95)°F.

c. At least once per 5 minutes in OPERATIONAL CONDITION 1 or 2 during testing which adds heat to the suppression chamber, by verifying the suppression chamber water temperature less than or equal to (105)°F.

d. At least once per 60 minutes when THERMAL POWER is less than or equal to (1)% of RATED THERMAL POWER and suppression chamber water temperature is greater than or equal to (95)°F, by verifying suppression chamber water temperature to be less than (110)°F.

1069 305

SURVEILLANCE REQUIREMENTS (Continued

e. At least once per 30 minutes following a scram from OPERATIONAL CONDITION 1 or 2 with the main steam line isolation valves closed and suppression chamber water temperature greater than or equal to (95)°F, by verifying suppression chamber water temperature less than (120)°F.

f. By an external visual examination of the suppression chamber after safety and safety/relief valve operation with the suppression chamber water temperature greater than or equal to (160)°F and reactor coolant system pressure greater than (200) psig.

g. At least once per 18 months by a visual inspection of the exposed accessible interior and exterior of the suppression chamber.

h. By verifying two suppression chamber water level instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 14 hours.

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

3. CHANNEL CALIBRATION at least once per 18 months.

1069 306

POOR ORIGINAL

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

(b. One OPERABLE RHR heat exchanger.)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within (72 hours) (7 days) or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool spray loops inoperable, (restore at least one loop to OPERABLE status within 8 hours or) be in at least HOT SHUTDOWN within (the next 12) (6) hours and in COLD SHUTDOWN* within the (following 24) (next 30) hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

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

b. By verifying that each of the required RHR pumps develops a flow of at least (300) gpm on recirculation flow through the RHR heat exchanger and suppression pool spray sparger when tested pursuant to Specification 4.0.5.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN within the time limit of this ACTION or as applicable, maintain reactor coolant temperature less than or equal to (400)°F by use of alternate heat removal methods.

1069 307

POOR ORIGINAL

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump; and

b. One OPERABLE RHR heat exchanger

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN* within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

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

b. By verifying that each of the required RHR pumps develops a flow of at least (7,700) gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN within the time limit of this ACTION or as applicable, maintain reactor coolant temperature less than or equal to (400)°F by use of alternate heat removal methods.

1069 308

POOR ORIGINAL

DRYWELL-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.2.4 Differential pressure between the drywell and suppression chamber shall be greater than or equal to (1._) psid* with two dry well-suppression chamber differential pressure instrumentation channels OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, during the period from:

a. Within 24 hours after THERMAL POWER is greater than (15%) of RATED THERMAL POWER following STARTUP, to

b. Within 24 hours prior to reducing THERMAL POWER to less than or equal to (15%) of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

a. With one drywell-suppression chamber differential pressure instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUT-DOWN within the following 24 hours.

b. With both drywell-suppression chamber differential pressure instrumentation channels inoperable, restore at least one inoperable channel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the drywell-suppression chamber differential pressure less than (1._) psid, restore the differential pressure to greater than or equal to (1._) psid within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.4.1 The drywell-suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.

4.6.2.4.2 Two drywell-suppression chamber differential pressure instrumentation channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 12 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 92 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

*Except for up to 4 hours for required surveillance which reduces the differential pressure.

1069 309

POOR ORIGINAL

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: As shown in Table 3.6.3-1.

ACTION:

a. With one or more of the containment isolation valves shown in Table 3.6.3-1 inoperable:

   1. Operation may continue provided that at least one isolation valve is maintained OPERABLE in each affected penetration that is open, and with 4 hours either; a) The inoperable valve(s) is restored to OPERABLE status or

      b) Each affected penetration is isolated by use of at least one deactivated automatic valve secured in the isolation position, or

      c) Each affected penetration is isolated by use of at least one closed manual valve or blind flange.

   2. Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   3. Otherwise, in OPERATIONAL CONDITION 4, 5, or *, suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel or a spent fuel shipping cask in the secondary containment, and with a potential for draining the reactor vessel. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. The provisions of Specification 3.0.3 are not applicable.

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable:

   1. Operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either;

      a) The inoperable valve is returned to OPERABLE status, or

      b) The instrument line is isolated and the associated instrument is declared inoperable.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*When handling irradiated fuel or a spent fuel shipping cask in the secondary containment.

1069 310

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow at greater than a (10) psid differential pressure.

1069 311

TABLE 3.6.3-1

CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	APPLICABLE OPERATIONAL CONDITIONS	ISOLATION TIME (Seconds)
a. Automatic Isolation Valves			

(a) See Specification 3.3.2, Table 3.3.2-1, (and Specification 3.3.6.1, Table 3.3.6.1-1) for isolation signal(s) that operates each valve group.
(b) May be opened on an intermittent basis under administrative control.
(c) Not subject to Type C leakage tests.
(d) Not a primary containment isolation valve. Listed for information only.
* But greater than or equal to (3) seconds.

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TABLE 3.6.3-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	APPLICABLE OPERATIONAL CONDITIONS
b. Manual Isolation Valves	
c. Other Isolation Valves	

(b) May be opened on an intermittent basis under administrative control.
(c) Not subject to Type C leakage tests.
(d) Excess flow check valve.

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Each pair of suppression chamber - drywell vacuum breakers shall be OPERABLE and in the fully closed position with:

a. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a (1) inch diameter orifice at a differential pressure of (1) psi,

b. The redundant position indicators OPERABLE, and

c. An opening setpoint of less than or equal to (0.5) psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more vacuum breakers in up to (three) pairs of suppression chamber - drywell vacuum breakers inoperable for opening but known to be in the closed position, the provisions of Specification 3.0.4 are not applicable and operation may continue until the next HOT SHUTDOWN provided the Surveillance Requirements of Specification 4.6.4.1.b.1 are performed on the OPERABLE pairs of vacuum breakers within 2 hours and at least once per 15 days thereafter. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one suppression chamber - drywell vacuum breaker in the open position, as indicated by the position indicating system, the provisions of Specification 3.0.4 are not applicable and operation may continue provided the Surveillance Requirements of Specification 4.6.4.1.b.1 are performed on the OPERABLE vacuum breakers and the other vacuum breaker in the pair is verified to be closed within 2 hours and at least once per 72 hours thereafter. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one position indicator of any suppression chamber - drywell vacuum breaker inoperable, the provisions of Specification 3.0.4 are not applicable and operation may continue provided the other vacuum breaker is verified to be closed within 2 hours and at least once per 15 days thereafter. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

1067 314

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. Within 2 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel and verifying that each breaker is closed as indicated by the position indication system.

2. At least once per 18 months by;

a) Verifying the opening setpoint, from the closed position, to be less than or equal to (0.5) psid,

b) Performance of a CHANNEL CALIBRATION which verifies that each position indicator indicates the associated vacuum breaker to be open if the vacuum breaker does not satisfy the delta P test in 4.6.4.1.b, and

c) Conducting a leak test at an initial differential pressure of (1) psi and verifying that the differential pressure does not decrease by more than (0.25) inches of water per minute for a (10) minute period.

1069 315

REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two Reactor Building - suppression chamber vacuum breakers shall be OPERABLE with an opening setpoint of less than or equal to (0.5) psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one Reactor Building - suppression chamber vacuum breaker inoperable for opening but known to be in the closed position, restore the inoperable vacuum breaker to OPERABLE status within (7) days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Reactor Building - suppression chamber vacuum breaker shall be demonstrated OPERABLE at least once per 18 months by:

a. Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of (0.5) psid, and

b. Visual inspection.

1069 316

3/4 6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours, or;

a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 5 or *, suspend handling of irradiated fuel and spent fuel shipping casks in the secondary containment, CORE ALTERATIONS and activities which could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 31 days:

1. All equipment hatches are closed and sealed, and
2. At least one door in each access to the Reactor Building is closed.

b. Verifying at least once per 92 days that each secondary containment automatic isolation damper is OPERABLE or secured in the closed position per Specification 3.6.5.2.

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to (1/4) inch of vacuum water gauge in less than or equal to (120) seconds, and
2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to (1/4) inch of vacuum water gauge in the secondary containment at a flow rate not exceeding (2300) CFM.

*When irradiated fuel or a spent fuel shipping cask is being handled in the secondary containment. SECONDARY CONTAINMENT INTEGRITY may be suspended as necessary to transfer a spent fuel shipping cask into and out of the secondary containment.

POOR ORIGINAL

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5 and *.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 inoperable:

a. Operation may continue provided that at least one isolation damper is maintained OPERABLE in each affected penetration that is open, and;

   1. The inoperable damper is restored to OPERABLE status within 8 hours, or

   2. The affected penetration is isolated by use of a closed damper or blind flange within 8 hours, or

   3. SECONDARY CONTAINMENT INTEGRITY is demonstrated within 8 hours and the valve is restored to OPERABLE status within 7 days.

b. Otherwise, in OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Otherwise, in OPERATIONAL CONDITION 5 or *, suspend handling of irradiated fuel and spent fuel shipping casks in the secondary containment, CORE ALTERATIONS and activities that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the damper to service after maintenance, repair or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifing the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation damper actuates to its isolation position within the isolation time.

*When irradiated fuel or a spent fuel shipping cask is being handled in the secondary containment.

1069 318

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

DAMPER FUNCTION	ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Supply Damper (_________)	(5)
2. Reactor Building Ventilation Supply Damper (_________)	(5)
3. Reactor Building Ventilation Exhaust Damper (_________)	(5)
4. Reactor Building Ventilation Exhaust Damper (_________)	(5)
5. ____________________	( )
6. ____________________	( )

1069 319

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5 and *.

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In OPERATIONAL CONDITION 5 or *, suspend all irradiated fuel and spent fuel shipping cask handling in the secondary containment, CORE ALTERATIONS or operations that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable in OPERATIONAL CONDITION 5 or *, suspend handling of irradiated fuel and spent fuel shipping casks in the secondary containment, CORE ALTERATIONS or operations that could reduce the SHUTDOWN MARGIN. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on automatic control.

*When irradiated fuel or a spent fuel shipping cask is being handled in the secondary containment.

SURVEILLANCE REQUIREMENTS (Continued)

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 subsystem by:

1. Verifying that the subsystem satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is (2300) cfm ± 10%.

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

3. Verifying a subsystem flow rate of (2300) cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than (8) inches Water Gauge while operating the filter train at a flow rate of (2300) cfm ± 10%

2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

   a. Plant exhaust plenum radiation - high,

   b. Drywell pressure - high,

   c. Reactor vessel water level - low, level 3, and

   d. Refueling floor exhaust radiation - high.

3. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

4. Verifying that the heaters dissipate (9.3) ± (1.0) kw when tested in accordance with ANSI N510-1975.

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SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to (99.95)*% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of (2300) cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than (99.95)*% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of (2300) cfm ± 10%.

*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99%, when a filter efficiency of 90% is assumed.

1069 322

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

PRIMARY CONTAINMENT HYDROGEN RECOMBINER SYSTEMS (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent primary containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum (heater sheath) temperature increases to greater than or equal to (700)°F within (90) minutes and is maintained for at least (2) hours.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

2. Verifying during a recombiner system functional test that the (heater sheath) temperature increases to greater than or equal to (1200)°F within (5) hours and is maintained between ( )°F and ( )°F for at least 4 hours.

3. Verifying the integrity of all heater electrical circuits by performing a continuity and resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to (10,000) ohms.

(4 Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners: i.e, loose wiring or structural connections, deposits of foreign materials, etc.)

1069 323

PRIMARY CONTAINMENT ATMOSPHERE DILUTION SYSTEM (If less than two hydrogen recombiners available)

LIMITING CONDITION FOR OPERATION

3.6.6.2 The primary containment atmosphere dilution (CAD) system shall be OPERABLE with:

a. An OPERABLE flow path capable of supplying nitrogen to the drywell, and

b. A minimum supply of (4350) gallons of liquid nitrogen.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the CAD system inoperable, restore the CAD system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The primary containment atmosphere dilution system shall be demonstrated to be OPERABLE:

a. At least once per 31 days by verifying that:

1. The system contains a minimum of (4350) gallons of liquid nitrogen, and

2. 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 18 months by:

1. Cycling each power operated (excluding automatic) valve in the flow path not testable during plant operation through at least one complete cycle of full travel, and

2. Verifying that each automatic valve in the flow path actuates to its correct position on a ____________ isolation test signal.

1069 324

PRIMARY CONTAINMENT HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 Two independent hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 Each hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days by:

1. Starting the system from the control room,
2. Verifying that the system operates for at least 15 minutes, and
3. Verifying that the system is aligned to receive electrical power from separate OPERABLE emergency buses.

b. At least once per 18 months by verifying a system flow rate of at least _______ cfm.

1069 325

PRIMARY CONTAINMENT OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.4 The primary containment atmosphere oxygen concentration shall be less than 4% by (volume).

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is gre: er than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdowr

ACTION:

With the oxygen concentration in the primary containment exceeding the limit, be in at least STARTUP within 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.4 The oxygen concentration in the primary containment shall be verified to be within the limit within 24 hours after THERMAL POWER greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

*See Special Test Exception 3.10.5.

1069 326

POOR ORIGINAL

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 Two independent residual heat removal service water (RHRSW) system subsystems shall be OPERABLE, with each subsystem comprised of:

a. Two OPERABLE RHRSW pumps, and

b. An OPERABLE flow path capable of taking suction from the (   ) and transferring the water through one RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one RHRSW pump in each subsystem inoperable, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one RHRSW subsystem inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With both RHRSW subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each residual heat removal service water system subsystem shall be demonstrated OPERABLE:

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

*Whenever two RHRSW subsystems are inoperable, if unable to attain COLD SHUTDOWN within the time limit of this ACTION or as applicable, maintain reactor coolant temperature less than or equal to (400)°F by use of alternate heat removal methods.

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SURVEILLANCE REQUIREMENTS (Continued)

b. By verifying the (   ) water level at the intake structure is greater than or equal to (   ) feet MSL;

   1. At least once per 14 days when the level is greater than (   ) feet MSL, and

   2. At least once per 12 hours when the level is less than or equal to (   ) feet MSL.

c. At least once per (12) months by verifying:

   1. The (   ) bottom conditions in the vicinity of the intake structure.

   2. The (   ) stage discharge rating curve in the unit vicinity.

1069 328

PLANT SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent plant service water system loops shall be OPERABLE with each loop comprised of:

a. Two OPERABLE plant service water pumps, and

b. An OPERABLE flow path capable of taking suction from the ( ) and transferring the water to (the associated safety related equipment).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one plant service water pump inoperable, restore the inoperable pump to OPERABLE status within 30 days or be in a least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one plant service water pump in each loop inoperable, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one plant service water system loop inoperable, restore the inoperable loop to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5, with up to three plant service water pumps and up to one loop inoperable, restore at least two pumps to OPERABLE status within 72 hours or declare the (associated safety related equipment) inoperable and take the ACTION required by Specifications (3.5.2 and 3.8.1.2).

SURVEILLANCE REQUIREMENTS

4.7.1.2 The plant service water system shall be demonstrated OPERABLE:

a. By verifying the ( ) water level at the intake structure is greater than or equal to ( ) feet MSL;

1. At least once per 14 days when the level is greater than ( ) feet MSL, and

2. At least once per 12 hours when the level is less than or equal to ( ) feet MSL.

SURVEILLANCE REQUIREMENTS (Continued)

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

c. At least once per 12 months by verifying:

   1. The (   ) bottom conditions in the vicinity of the intake structure.

   2. The (   ) stage discharge rating curve in the unit vicinity.

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

   1. Each automatic valve servicing non-safety related equipment actuates to its isolation position on an isolation test signal.

   2. Each pump starts automatically, when on Standby, to maintain service water pressure greater than or equal to (60) psig.

1069 330

ULTIMATE HEAT SINK (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.7.1.3 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation (   ) Mean Sea Level, USGS datum, and

b. An average water temperature of less than or equal to (   )°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water temperature and water level to be within their limits.

1067 331

CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency filtration system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 5 and *.

ACTION:

With one control room emergency filtration subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or:

a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 5 or *, within the next 6 hours initiate and maintain operation of the control room emergency filtration system in the (pressurization) mode of operation.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency filtration subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to (120)°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters on (automatic control).

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

   1. Verifying that with the subsystem operating at a flow rate of (2000) cfm ± 10% and exhausting through the HEPA filters and charocal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the subsystem diverting valve, is less than or equal to 1% when the subsystem is tested by admitting cold DOP at the system intake.

*When irradiated fuel or a spent fuel shipping cask is being handled in the secondary containment.

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SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the subsystem satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is (2000) cfm ± 10%.

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

4. Verifying a subsystem flow rate of (2000) cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the subsystem at a flow rate of (2000) cfm ± 10%.

2. Verifying that on each of the below (isolation) mode actuation test signals, the subsystem automatically switches to the (isolation) mode of operation and the isolation valves close within ( ) seconds:

a) ______________,
b) ______________,
c) ______________, and
d) ______________.

3. Verifying that on each of the below (pressurization) mode actuation test signals, the subsystem automatically switches to the (pressurization) mode of operation and the control room is maintained at a positive pressure (of (1/8 inch W.G.) relative to the outside atmosphere during subsystem operation at a flow rate less than or equal to ( ) cfm:

a) ______________,
b) ______________, and
c) ______________.

1069 333

4. Verifying that the heaters dissipate (7.5) ± (0.75) Kw when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to (99.95)%* of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of (2000) cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove (99.95)%* of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of (2000) cfm ± 10%.

*99.95% applicable when a filter efficiency of 99% is assumed in the safety analyses; 99%, when a filter efficiency of 90% is assumed.

1069 334

3/4.7.3 FLOOD PROTECTION (OPTIONAL*)

LIMITING CONDITION FOR OPERATION

3.7.3 Flood protection shall be provided for all safety related systems, components and structures when the water level of the ( ) exceeds ( ) Mean Sea Level USGS datum at ( ).

APPLICABILITY: At all times.

ACTION:

With the water level at ( ) above elevation ( ) Mean Sea Level USGS datum:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

b. Initiate and complete within ( ) hours the following flood protection measures:

1. ________________________________,

2. ________________________________, and

3. ________________________________.

SURVEILLANCE REQUIREMENTS

4.7.3 The water level at ( ) shall be determined to be within the limit by:

a. Measurement at least once per 24 hours when the water level is below elevation ( ) Mean Sea Level USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation ( ) Mean Sea Level USGS datum.

*This specification not required if the facility design has adequate passive flood control protection features sufficient to accommodate the Design Basis Flood identified in Regulatory Guide 1.59, August 1973.

1069 335

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of (automatically) taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than (113) psig.

ACTION:

With the RCIC system inoperable, operation may continue and the provisions of Specification 3.0.4 are not applicable provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within (14) days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to (113) psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water, and

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

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to (400) gpm in a test flow path when steam is being supplied to the turbine at normal reactor vessel operating pressure, (1000 + 20, - 80) psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.

2. Verifying that the system will develop a flow of greater than or equal to (400) gpm on recirculation flow when steam is supplied to the turbine at a pressure of (150) ± (15) psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal and on a suppression pool high water level signal.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

1069 337

3/4.7.5 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.5 All hydraulic snubbers shown in Table 3.7.5-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one or more hydraulic snubbers inoperable, replace or restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 Hydraulic snubbers shown in Table 3.7.5-1 shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.7.5.2 Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7.5-1 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7.5-1 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

4.7.5.3 Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

4.7.5.4 At least once per 18 months during shutdown, a representative sample of at least 10 snubbers or at least 10% of all snubbers listed in Table 3.7.5-1, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbbers greater than 50,000 pound capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis except snubbers identified in Table 3.7.5-1 as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

TABLE 3.7.5-1

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

*Snubbers may be added to safety related systems without prior License Amendment to Table 3.7.5-1 provided that a revision to Table 3.7.5-1 is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7.5-1 is included with the next License Amendment Request.

TABLE 4.7.5-1

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
≥ 8	31 days ± 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

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

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3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. 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

4.7.6.1 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.

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

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

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

   2. In any form other than gas.

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

1069 342

3/4.7.7 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The fire suppression water system shall be OPERABLE with;

a. (Two) OPERABLE fire suppression (diesel driven) fire pumps, each with a capacity of (2500) gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies, each with a minimum contained volume of _____ gallons, and

c. An OPERABLE flow path capable of taking suction from the _____ tank and the ____ tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.7.2, 3.7.7.5, and 3.7.7.6.

APPLICABILITY: At all times.

ACTION:

a. With one fire pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply. 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,

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.

1069 343

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

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

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven fire suppression pump and operating it for at least 15 minutes on recirculation flow.

c. 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.

d. (At least once per 6 months by performance of a system flush.)

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position,

   2. Verifying that each fire suppression pump develops at least (2500) gpm at a system head of (250) feet,

   3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to ___ psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.7.1.2 Each fire suppression pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1. The fuel day tank contains at least ( ) gallons of fuel.

   2. The fuel storage tank contains at least ( ) gallons of fuel.

   3. The fuel transfer pump starts and transfers fuel from the storage tank to the day tank, and

   4. The diesel starts from ambient conditions and operates for greater than or equal to 30 minutes on recirculation flow.

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SURVEILLANCE REQUIREMENTS (Continued)

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 D975-74 when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.7.1.3 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 pilot cell is above the plates,
2. The pilot cell specific gravity, corrected to (77)°F and full electrolyte level, is greater than or equal to (1.200),
3. The pilot cell voltage is greater than or equal to (24) volts, and
4. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to (24) volts under float charge and has not decreased more than (2.0) volts from the value observed during the original acceptance test,

2. The specific gravity, correted to (77)°F and full electrolyte level, of each connected cell is greater than or equal to ( ) and has not decreased more than ( ) from the value observed during the revious test, and

3. The electrolyte level of each connected cell is above the plates.

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

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

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

1069 345

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.2 The following spray and/or sprinkler systems shall be OPERABLE:

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, without one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant 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, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.7.2 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 is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a ____________ test signal, and

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SURVEILLANCE REQUIREMENTS (Continued)

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each (deluge) nozzle's spray area to verify that the spray pattern is not c structed.

d. At least once per 3 years by performing ar air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

1069 347

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.3 The following low pressure and high pressure CO_2 systems shall be OPERABLE.

a. (Plant dependent - to be listed by name and location.)

b.

c.

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant 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, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.7.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE 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.

4.7.7.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than ______ and pressure to be greater than _____ psig, and

b. At least once per 18 months by verifying:

1. The system valves and associated ventilation dampers and fire door release mechanisms actuate manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

1069 348

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.3.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying the CO_2 storage tank weight to be at least 90% of full charge weight.

b. At least once per 18 months by:

1. Verifying the system and associated ventilation dampers and fire door release mechanisms actuation manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

1069 349

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. (Plant dependent - to be listed by name and location.)

b.

c.

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

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant 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, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and 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

4.7.7.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

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

b. At least once per 18 months by:

1. Verifying the system and associated ventilation dampers and fire door release mechanisms actuation manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

1069 350

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.7.5 The fire hose stations shown in Table 3.7.7.5-1 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 3.7.7.5-1 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.7.5 Each of the fire hose stations shown in Table 3.7.7.5-1 shall be demonstrated OPERABLE:

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

b. At least once per 18 months by:

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

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

c. At least once per 3 years by:

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

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any hose station.

TABLE 3.7.7.5-1

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE RACK INDENTIFICATION

*List all fire hose stations required to ensure the OPERABILITY of safety related equipment.

1069 352

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.7.6 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.6-1 shall be OPERABLE.

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

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Tabl 3.7.7.6-1 inoperable, route 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) within one hour if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, route an additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.7.6 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.6-1 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, during March, April or May and 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. Replacement of all degraded gaskets in couplings.

1069 353

TABLE 3.7.7.6-1

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION*	HYDRANT NUMBER

*List all yard fire hydrants and hydrant hose houses required to ensure the OPERABILITY of safety related equipment.

1069 354

3/4.7.8 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.8 All fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers, in fire zone boundaries protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour 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 non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetrations(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

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

SURVEILLANCE REQUIREMENTS

4.7.8 Each of the above required penetration fire barriers shall be verified to be functional:

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

b. Prior to restoring a fire barrier penetration to functional status following repairs or maintenance, by performance of a visual inspection of the affected fire barrier penetration.

1069 355

3/4.7.9 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7.9-1 shall be maintained within the limits indicated in Table 3.7.9-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceed the temperature limit(s) for Equipment Not Operating shown in Table 3.7.9-1 for more than eight hours, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission purusuant to Specification 6.9.2 within the next 30 days providing a record of the cumulative time the temperature in the affected area exceeded its limit.

b. With a temperature exceeding the limit for Equipment Operating shown in Table 3.7.9-1, in addition to submitting a Special Report as described in ACTION a. above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

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

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas shown in Table 3.7.9-1 shall be determined to be within its limit at least once per 24 hours.

1069 356

TABLE 3.7.9-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)	
	EQUIPMENT NOT OPERATING	EQUIPMENT OPERATING
a.		
b.		
c.		
d.		
e.		

1069 357

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators each with:

   1. Separate day and engine mounted fuel tanks containing a minimum of (250) gallons of fuel,

   2. A separate fuel storage system containing a minimum of (26,000) gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator (1A) or (1B) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, within one hour and at least once per 8 hours therafter; restore at least two offsite circuits and diesel generators (1A) and (1B) to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one offsite circuit and diesel generator (1A) or (1B) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, within one hour and at least once per eight hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators (1A) and (1B) to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

POOR ORIGINAL

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With diesel generators (1A) and (1B) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators (1A) and (1B) to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators (1A) and (1B) to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With diesel generator (1C) of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore the inoperable diesel generator (1C) to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

1069 359

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day and engine-mounted fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day and engine-mounted fuel tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least (900) rpm in less than or equal to (10) seconds. The generator voltage and frequency shall be _____ ± _____ volts and _____ ± _____ Hz within _____ seconds after the start signal. The diesel generator shall be started for this test by using the following signals which shall be changed on a rotating basis:

a) Manual.

b) Simulated loss of offsite power by itself.

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

d) An ESF actuation test signal by itself.

5. Verifying the diesel generator is synchronized, loaded to greater than or equal to (continuous rating) kw in less than or equal to (10) seconds, and operates for greater than or equal to 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to (140) psig.)

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SURVEILLANCE REQUIREMENTS (Continued)

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-D975-74 when checked for viscosity, water and sediment.

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

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

2. Verifying the diesel generator capability to reject a load of greater than or equal to (largest single emergency load) kw for diesel generator (1A), greater than or equal to (largest single emergency load) kw for diesel generator (1B), and greater than or equal to (largest single emergency load) kw for diesel generator (1C) while maintaining voltage at (4160) volts ± (10)% and frequency at (60) Hz ± (2)%.

3. Verifying the diesel generator capability to reject a load of (continuous rating) kw without exceeding 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is lower. The generator voltage shall not exceed ___ volts during and following the load rejection.

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

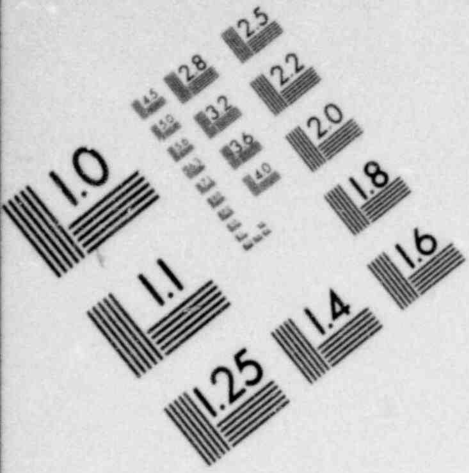
a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within ___ seconds, energizes the auto-connected shutdown loads through the load sequence and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the voltage and frequency of the emergency busses shall be maintained at _____ ± _____ volts and _____ ± _____ Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be _____ ± _____ volts and _____ ± _____ Hz within _____ seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

1069 361

7



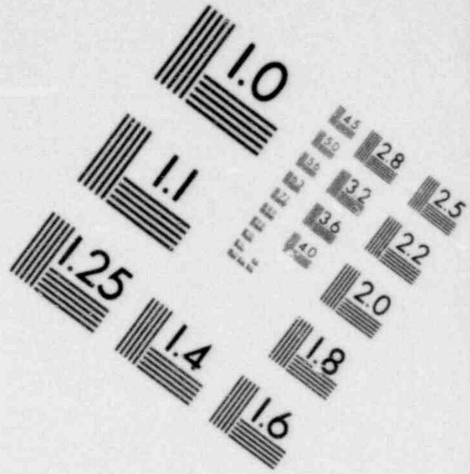
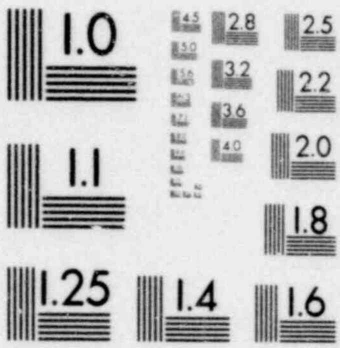


IMAGE EVALUATION
TEST TARGET (MT-3)



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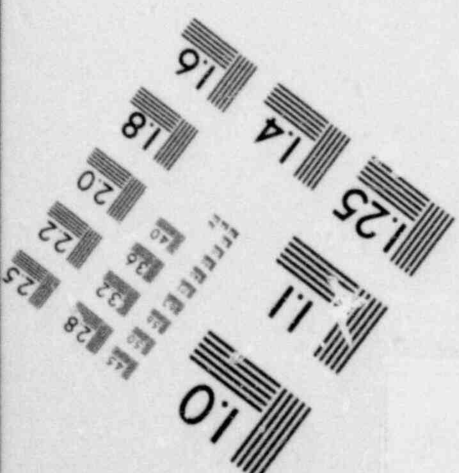

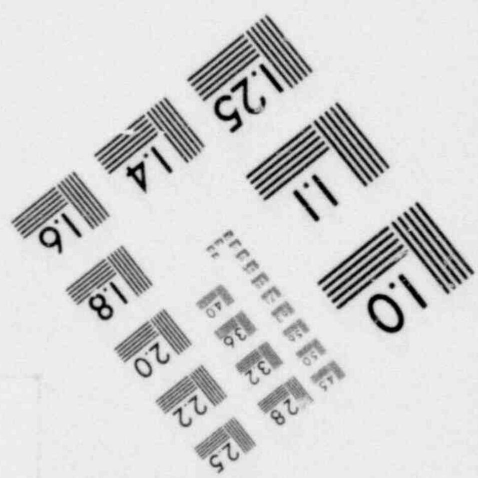

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

6. Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of offsite power in conjunction with an (ECCS) actuation test signal, and

   a) Verifying deenergization of the emergency busses and loads shedding from the emergency busses.

   b) Verifying the diesel generator starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connnected loads, energizes the auto-connected emergency (accident) loads within _____ seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while the load its generator is loaded with the emergency loads. After energization, the voltage and frequency of the emergency busses shall be maintained at _____ ± _____ volts and _____ ± _____ Hz during this test.

   c) Verifying that all automatic diesel generator trips, except (engine overspeed and generator differential), are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to (2-hour rating) kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to (continuous rating) kw. Within 5 minutes after completing this 24-hour test, repeat Specification 4.8.1.1.2.c.4. The generator voltage and frequency shall be ______ ± volts and _____ ± generator voltage and frequency shall be maintained within these limits during this test.

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

10. Verifying the diesel generator's capability to:

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

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

    c) Be restored to its standby status.

SURVEILLANCE REQUIREMENTS (Continued)

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

12. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day and engine-mounted tank of each diesel via the installed cross connection lines.

13. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

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

    a) __________
    b) __________

15. Verifying that with all diesel generator air start receivers pressurized to less than or equal to (   ) psig and the compressors secured, the diesel generator starts at least ( ) times from ambient conditions and accelerates to at least (900) rpm in less than or equal to (10) seconds.)

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

4.8.1.1.3 The starting (and control power) battery and battery charger of each diesel generator shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks.

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, of each cell is greater than or equal to ( ),

3. The pilot cell voltage is greater than or equal to (   ) volts, and

4. The overall battery voltage is greater than or equal to ( ) volts.

1070 002

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to ( ) volts under float charge and has not decreased more than ( ) volts from the value observed during the original acceptance test,

2. The specific gravity, corrected to (77)°F and full electrolyte level, of each cell is greater than or equal to ( ) and has not decreased more than 0.04 from the value observed during the previous test, and

3. The electrolyte level of each cell is between the minimum and maximum level indication marks.

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

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

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

3. The battery charger will supply at least ( ) amperes at a minimum of ( ) volts for at least ( ) hours.

d. At least once per 60 months during shutdown by verifying that the battery capacity is at least 90% of the manufacturers rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

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

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31 day test frequency.

1070 004

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator (1A) or (1B), and diesel generator (1C) when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. Day and engine mounted fuel tanks containing a minimum of (250) gallons of fuel.

   2. A fuel storage system containing a minimum of (31,000) gallons of fuel.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With all offsite circuits inoperable and/or with diesel generators (1A) and (1B) inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel or a spent fuel shipping cask in the secondary containment and all operations that have a potential for draining the reactor vessel. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY with 8 hours.

b. With diesel generator (1C) inoperable, restore the inoperable diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1, 4.8.1.1.2, 4.8.1.1.3 and 4.8.1.1.4, except for the requirement of 4.8.1.1.2.a.5.

*When handling irradiated fuel or a spent fuel shipping cask in the secondary containment.

1070 005

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A. C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. distribution system electrical divisions shall be OPERABLE:

a. Division (I), consisting of:

1. 4160 volt bus (____).
2. 480 volt buses (____ and _____).
3. 120 volt A.C. distribution panels in (480 volt MCCs _____, _____, _____ and _____).

b. Division (II), consisting of:

1. 4160 volt bus (_____).
2. 480 volt buses (_____ and _____).
3. 120 volt A.C. distribution panels in (480 volt MCCs _____, _____, _____ and _____).

c. Division (III), consisting of:

1. 4160 volt bus (_____).
2. 480 volt MCC (_____).
3. 120 volt A.C. distribution panels in (480 volt MCC _____).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division (I) or Division (II) of the above required A.C distribution system inoperable, restore the inoperable division to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division (III) of the above required A.C. distribution system inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The above required A.C. distribution system electrical divisions shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

1070 006

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division (I) or Division (II), and when the HPCS system is required to be OPERABLE Division (III) of the A.C. distribution system shall be OPERABLE with:

a. Division (I), consisting of:

1. 4160 volt bus (_____).
2. 480 volt buses (_____ and _____).
3. 120 volt A.C. distribution panels in (480 volt MCCs _____, _____, _____ and _____).

b. Division (II), consisting of:

1. 4160 volt bus (_____).
2. 480 volt buses (_____ and _____).
3. 120 volt A.C. distribution panels in (480 volt MCCs _____, _____, _____ and _____).

c. Division (III), consisting of:

1. 4160 volt bus (_____).
2. 480 volt MCC (_____).
3. 120 volt A.C. distribution panels in (480 volt MCC_____).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With both Division (I) and (II) of the above required A.C. distribution system inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel or spent fuel shipping cask in the secondary containment, and all operations that have a potential for draining the reactor vessel. In OPERATIONAL CONDITION 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

b. With Division (III) of the above required A.C. distribution system inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required A.C. distribution system electrical division(s) shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

*When handling irradiated fuel or a spent fuel shipping cask in the secondary containment.

1070 007

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. distribution system electrical divisions shall be OPERABLE:

a. Division (I), consisting of:
   1. 125 volt battery (1A).
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel (_____).

b. Division (II), consisting of:
   1. 125 volt battery (1B).
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel (_____).

c. Division (III), consisting of:
   1. 125 volt battery (1C).
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel (_____).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division (I) or Division (II) inoperable, restore the inoperable division to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division (III) inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each of the above required D.C. distribution system electrical divisions shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability with an overall voltage greater than or equal to (250/125) volts.

1070 008

SURVEILLANCE REQUIREMENTS (Continued)

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

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is greater than or equal to (   ), and

3. The pilot cell voltage is greater than or equal to (   ) volts.

b. At least once per 92 days by verifying that:

1. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

2. The average specific gravity of all connected cells, corrected to 77°F and full electrolyte level,   greater than or equal to (   ).

3. The electrolyte temperatures in a representative sample of cells consisting of at least every si th cell are within ± 5°F.

4. The minimum specific gravity, corrected to (77°)F and full electrolyte level, of each connected cell is within 0.010 of the average corrected value of all the connected cells.

5. The voltage of each connected cell is within ± 0.04 volts of the average voltage of all the connected cells,

6. The total battery terminal voltage is greater than or equal to _________ volts, and

7. The battery load (charger current) with the battery on float charge is less than _________ amps.

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

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

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

3. The resistance of each cell-to-cell and terminal connection is less than or equal to (   ) ohms, and

4. The battery charger will supply at least (   ) amperes at a minimum of (   ) volts for at least (4) hours.

1070 009

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

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for (8) hours when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage less than or equal to ( ) volts.

a) Battery (1A), greater than or equal to ( ) amperes; battery (1B), greater than or equal to ( ) amperes; and battery (1C), greater than or equal to ( ) amperes during the initial 60 seconds of the test.

b) Battery (1A), greater than or equal to ( ) amperes; battery (1B), greater than or equal to ( ) amperes; and battery (1C), greater than or equal to ( ) amperes during the remainder of the first hour of the test.

c) Battery (1A), greater than or equal to ( ) amperes; battery (1B), greater than or equal to ( ) amperes; and battery (1C), greater than or equal to ( ) amperes during the remainder of the (8) hour test.

3. At the completion of either of the above tests, the battery charger shall be demonstrated capable of recharging its battery at a rate of at least ( ) amperes for charger (1A), ( ) amperes for charger (1B), and ( ) amperes for charger (1C) while supplying normal D.C. loads. The battery shall be charged to at least 95% capacity in less than or equal to 24 hours.

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 90% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, Division (I) or Division (II), and, when the HPCS system is required to be OPERABLE, Division (III) of the D.C. distribution system shall be OPERABLE with:

a. Division (I), consisting of:
   1. 125 volt battery (1A).
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel (____).

b. Division (II), consisting of:
   1. 125 volt battery (1B).
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel (____).

c. Division (III), consisting of:
   1. 125 volt battery (1C).
   2. 125 volt full capacity charger.
   3. 125 volt distribution panel (____).

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With both Division (I) and Division (II) of the above required D.C. distribution system inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel or a spent fuel shipping casks in the secondary containment, and all operations that have a potential for draining the reactor vessel.

b. With Division (III) of the above required D.C. distribution system inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 At least the above required D.C. distribition system electrical division(s) shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability with an overall voltage of greater than or equal to (250/125) volts.

4.8.2.4.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

*When handling irradiated fuel or a spent fuel shipping cask in the secondary containment.

1070 011

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following A.C. circuits inside primary containment shall be de-energized*:

a. Circuit numbers (__, __, __ and __) in panel (   ).

b. Circuit numbers (__, __, __ and __) in panel (   ).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required A.C. circuits shall be determined to be de-energized at least once per 24 hours by verifying that the associated circuit breakers in the specified panels are in the tripped condition.

*Except during entry into the drywell

1070 012

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.2 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.3.2-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.3.2-1 inoperable:

a. De-energize the circuit(s) by tripping the associated (backup) circuit breaker(s) within 72 hours and verify the circuit breaker to be tripped at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped; or

b. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.2 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.3.2-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level and performing:

   a) A CHANNEL CALIBRATION of the associated protective relays, and

   b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8.3.2-1.

   For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

1070 013

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

1070 014

TABLE 3.8.3.2-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER* AND LOCATION	TRIP SETPOINT (Amperes)	RESPONSE TIME (Milliseconds/Cycles)	SYSTEM/ COMPONENT POWERED
a. ( ) KV Circuit Breakers			
b. ( ) KV Circuit Breakers			
c. (Molded Case) Circuit Breakers			
1. Type ______			
		(N.A.)	
		(N.A.)	
		(N.A.)	
2. Type ______			
		(N.A.)	
		(N.A.)	
		(N.A.)	

*List all primary and backup breakers.

1070 015

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.3 The thermal overload protection and/or bypass devices, integral with the motor starter, of each valve listed in Table 3.8.3.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s).

SURVEILLANCE REQUIREMENTS

4.8.3.3 The above required thermal overload protection and/or bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

 1. Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

 2. Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

1070 016

TABLE 3.8.3.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Yes/No)

1070 017

3/4.9.1 REACTOR MODE SWITCH

POOR ORIGINAL

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position, except that when the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be inserted or withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Fuel grapple position.
5. Service platform hoist fuel-loaded.
6. Source range monitor countrate.

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

d. The provisions of Specification 3.0.3 are not applicable.

---

* See Special Test Exceptions 3.10.1 and 3.10.3.

The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever the reactor pressure vessel head is unbolted or removed and fuel is in the vessel.

1070 018

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

a. Within 2 hours prior to:

1. Beginning CORE ALTERATIONS, and

2. Resuming CORE ALTERATIONS when the reactor mode switch has been unlocked.

b. At least once per 12 hours.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks shall be demonstrated OPERABLE:

a. Within 24 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS, and

b. Prior to resuming CORE ALTERATIONS following repair, maintenance or replacement of any component that could affect the Refuel position interlocks.

1070 019

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level:

a. Each with continuous visual indication in the control room,

b. At least one with audible indication in the control room and on the refueling floor,

c. One of the SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and another SRM detector located in an adjacent quadrant, and

d. The "shorting links" removed from the RPS circuitry during CORE ALTERATIONS and shutdown margin demonstrations.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS** and fully insert all insertable control rods. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

**Except movement of IRM, SRM or special movable detectors.

1070 020

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps:

1. Prior to control rod withdrawal,

2. At least once per 12 hours during CORE ALTERATIONS, and

3. At least once per 24 hours.

d. Verifying that the RPS circuitry "shorting li   have been removed and that the RPS circuitry is in a non-coincidence trip mode within 8 hours prior to starting CORE ALTERATIONS or shutdown margin demonstrations.

1070 021

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be fully inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

a. With all control rods not fully inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be fully inserted, except as above specified:

a. Within 2 hours prior to:

1. The start of CORE ALTERATIONS.

2. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.

b. At least once per 12 hours.

* Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

1070 022

3/4.9.4 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor shall be subcritical for at least (24) hours.

APPLICABILITY: OPERATIONAL CONDITION 5, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than (24) hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least (24) hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

1070 023

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling platform personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

1070 024

3/4.9.6 REFUELING PLATFORM OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling platform crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff when the load exceeds (2000) pounds.

b. Demonstrating operation of the loaded interlock when the load exceeds (1000) pounds.

c. Demonstrating operation of the downtravel stop when downtravel exceeds (___) feet.

d. Demonstrating operation of the up-travel stop when up-travel exceeds (___) feet.

e. Demonstrating operation of the slack cable cutoff when the load is less than (100) pounds.

f. Performing a load test of at least (_____) pounds.

1070 025

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of (2500) pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of (2500) pounds over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.

1070 026

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least (23) feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITION 5, during handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

1070 027

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least (23) feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

1070 028

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown or Refuel position per Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are fully inserted and electrically or hydraulically disarmed.

e. All other control rods are fully inserted.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the reactor pressure vessel and initiate action to satisfy the above requirements. The provisions of Specification 3.0.3 are not applicable.

1070 029

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown or Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1,

b. The SRM channels are OPERABLE per Specification 3.9.2,

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied as above specified,

d. All other control rods in a five-by-five array centered on the control rod being removed are fully inserted and electrically or hydraulically disarmed, and

e. All other control rods are fully inserted.

1070 030

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core.

a. The reactor mode switch is locked in the Shutdown or Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed.

e. The four fuel assemblies are removed from the core cell surrounding each control rod or rod drive mechanism to be removed.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the reactor pressure vessel and initiate action to satisfy the above requirements. The provisions of Specification 3.0.3 are not applicable.

1070 031

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown or Refuel position per Specification 3.9.1,

b. The SRM channels are OPERABLE per Specification 3.9.2,

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied,

d. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism that is to be removed from the reactor vessel at the same time are removed from the core.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

1070 032

3/4.9.11 REACTOR COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.11 The shutdown cooling mode of the residual heat removal (RHR) system shall be OPERABLE with at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the shutdown cooling mode of the RHR system inoperable, suspend all operations involving an increase in the reactor decay heat load. Close all secondary containment penetrations providing direct access from the secondary containment atmosphere to the outside atmosphere within 4 hours.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The shutdown cooling mode of the RHR system shall be determined OPERABLE at least once per 31 days by:

a. Starting at least one RHR pump from the control room, if not already operating, and

b. Verifying that system valves are properly aligned to provide recirculation of reactor coolant through the RHR heat exchanger.

1070 033

SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed, the primary containment air lock doors to be open and the reactor mode switch to be in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than (1)% of RATED THERMAL POWER and reactor coolant temperature less than 212°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER exceeding greater than or equal to (1)% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 212°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

1070 034

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM (Group Notch Type)

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of the individual rod position bypass switches for the following tests provided that at least the requirements of Specifications 3.1.3.1 and 3.1.4.1 are satisfied:

a. Shutdown margin demonstrations, Specification 4.1.1,

b. Control rod scram, Specification 4.1.3.2,

c. Control rod friction measurements, and

d. Startup Test Program with the THERMAL POWER less than (20)% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RSCS are bypassed, verify;

a. That the RWM is OPERABLE per Specification 3.1.4.1,

b. That movement of the control rods from (50)% ROD DENSITY to the RSCS preset power level is blocked or limited to the single notch mode, and

c. Conformance with this specification and procedures by a second licensed operator or other qualified member of the technical staff.

1070 035

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM (Banked Position Type)

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of (bypassing the low power setpoint switches) for the following tests provided that at least the requirements of Specification 3.1.3.1 and 3.1.4.1 are satisfied:

a. Shutdown margin demonstrations, Specification 4.1.1,

b. Control rod scram, Specification 4.1.3.2,

c. Control rod friction measurements, and

d. Startup Test Program with the THERMAL POWER less than (20)% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RSCS are bypassed, verify;

a. That the RWM is OPERABLE per Specification 3.1.4.1,

b. That movement of the control rods from (75)% ROD DENSITY to the RSCS preset power level is blocked or limited to the banked position mode, and

c. Conformance with this specification and procedures by a second licensed operator or other qualified member of the technical staff.

1070 036

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3 and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that the following requirements are satisfied.

1. The source range monitors are OPERABLE with the RPS circuitry shorting links removed per Specification 3.9.2.
2. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other qualified member of the technical staff.
3. The "rod-out-notch-override" control shall not be used during movement of the control rods.
4. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other qualified member of the technical staff is present to verify compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

1070 037

3/4.10.4 RFCT''CULATION L(PS

LIMIT:NG C(ION FOR OPERAT'ON

3.10.4 The requi'ements of Specification 3.4.1.1 that recirculation loops be in operation may be suspended for up to 24 hc rs for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed (5)% of RATED THERMAL POWER, and

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, immediately fully insert all control rods.

b. With the above specified THERMAL POWER limit exceeded, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than (5)% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

1070 038

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.6 may be suspended during the performance of the Startup Test Program until either the required 100% of RATED THERMAL POWER trip tests have been completed or the reactor has operated for 120 Effective Full Power Days.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The Effective Full Power Days of operation shall be verified to be less than 120, by calculation, at least once per 7 days during the Startup Test Program.

1070 039

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one LPCI subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 212°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

1070 040

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

1070 041

NOTE

The Summary statements contained in this section provide the bases for the specifications in Section 3.0 and 4.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 calls for two control room emergency filtration subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the terms of Specification 3.0.3, if both of the required subsystems are inoperable, the unit is to be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the next 30 hours. As a further example, Specification 3.6.6.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the terms of Specification 3.0.3, if both of the required systems are inoperable, the unit is to be in at least HOT SHUTDOWN within 6 hours. The unit shall be brought to the required OPERATIONAL CONDITION within the required times by promptly initiating and carrying out an orderly shutdown. It is intended that this guidance also apply whenever an ACTION statement requires a unit to be in STARTUP within 2 hours or in HOT SHUTDOWN within 6 hours.

3.0.4 This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS or other specified applicability condition for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL CONDITIONS or other specified applicability condition are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance; instead, it permits the more frequent performance of surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that surveillance activities associated with a Limiting Conditions for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

1070 044

APPLICABILITY

BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

1070 045

POOR ORIGINAL

3/4.1.1 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.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + (0.38)% delta K or R + (0.28)% delta K, as appropriate. The value of R in units of % delta K is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

1070 046

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MPCR from becoming less than (1.07) during the limiting power transient analyzed in Section (15.__) of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MPCR remains greater than (1.07). The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactors.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than (3) inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section (15.__) of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

1070 048

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for (maintaining the reactor in the subcritical condition as the plant cools to ambient in the vent insufficient control mode are inserted) (bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that (none) of the withdrawn control rods can be inserted). To meet this objective it is necessary to inject a quantity of boron which produces a concentration of (600) ppm in the reactor core in approximately (90 to 120) minutes. A normal quantity of (3470) gallons of solution having a (13.7)% sodium pentaborate concentration is required to meet a shutdown requirement of (3)%. There is an additional allowance of (150) ppm in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is (41.2) gpm. The maximum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are estalbished on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

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1. C. J. Paone, R. C. Stirn and J. A. Woodley, "Rod Drop Accident analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972
2. C. J. Paone, R. C. Stirn and R. M. Young, Supplement 1 to NESO-10527, July 1972
3. J. A. Haum, C. J. Paone and R. C. Stirn, addendum 2 "Exposed Cores" supple ment 2 to NEDO-10527, January 1973

1070 049

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in the Final Acceptance Criteria (FAC) issued in January, 1974, considering the postulated effects of fuel pellet densification.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times (1.02) is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures (3.2.1-1. 3.2.1-2 and 3.2.1-3).

The calculational procedure used to establish the APLHGR shown on Figures (3.2.1-1, 3.2.1-2 and 3.2.1-3) is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

1070 050

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a TOTAL PEAKING FACTOR of (2.43) for (8 x 8) fuel. The flow biased simulated thermal power-high scram setting and flow biased simulated thermal power-high control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.0 in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of power and peak flux indicates a TOTAL PEAKING FACTOR greater than (2.43). The method used to determine the design TPF shall be consistent with the method used to determine the MTPF.

1070 051

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... (2535) Mwt which corresponds to (105)% of licensed core power*

Vessel Steam Output .... (11) x 10^6 lbm/h which corresponds to (105)% of rated steam flow

Vessel Steam Dome Pressure.............. (1055) psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks (2.2) ft^2, (1.7) ft^2, (1.3) ft^2

b. Small Breaks (1.0) ft^2, (0.8) ft^2, (0.07) ft^2, (0.05) ft^2, and (0.02) ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	(13.4)	(1.4)	(1.18)

A more detailed listed of input of each model and its source is presented in Section II of Reference 1 and subsection (15._) of the FSAR.

*This power level meets the Appendix requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at (102)% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

1070 052

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of (1.07), and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of (1.07), the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table (15._) that are input to a GE-core dynamic behavior transient computer program described in NEDO-10802[3]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[1] The principal result of this evaluation is the reduction in MCPR cause by the transient.

The purpose of the K_f factor of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the operating limit MCPR and the K_f factor. The K_f factors assure that the operating limit MCPR of Specification 3.2.3 will not be violated. The K_f factors were derived using THERMAL POWER and core flow corresponding to (105)% of rated steam flow.

The K_f factors were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the (105)% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the (105)% of rated steam flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

1070 053

POOR ORIGINAL

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to (25)% of RATED THERMAL POWER, the reactor will be operating at (minimum) recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at (25)% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to (25)% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. (The power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a (95)% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.)

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE Bwr, February 1973 (NEDO-10802).

1070 054

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

1070 055

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a (3) second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of (13) seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the (13) second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the (13) second delay. It follows that checking the valve speeds and the (13) second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

1070 056

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, and Appendix (___) of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

1070 057

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. (This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.)

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. (This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.)

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintence of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost (and is consistent with General Design Criteria 19 of 10 CFR 50.)

1070 058

3/4.3.7.5 POST-ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975.)

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector to be used and determining the acceptability of its voltage curve. For the purpose of measuring _______ a full incore flux map is used. Quarter-core flux maps, as defined in _______, may be used and full incore flux maps or symmetric incore thimbles may be used for measuring _________.

3/4.3.7.8 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection system ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provided protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the (isolation) mode of operation to provide the required protection. (The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", February 1975.)

3/4.3.7.9 CHLORIDE INTRUSION MONITORS

The chloride intrusion monitors provide adequate warning of any leakage in the condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. With only a minimum number of instruments available increased sampling frequency provides adequate information for the same purpose.

1070 059

3/4.3.7.10 FIRE DETECTION INSTRUMENTATION

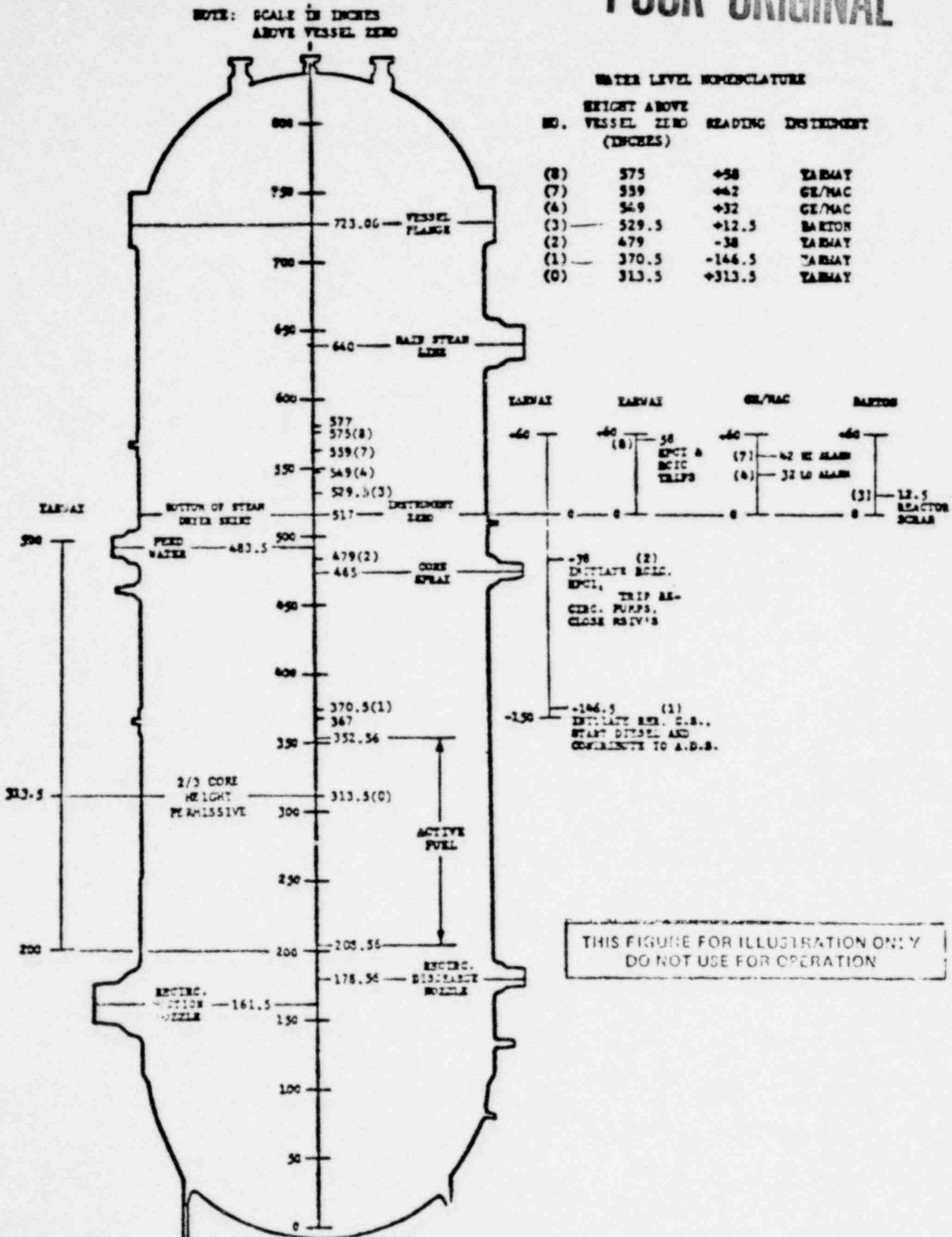
OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

POOR ORIGINAL



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON
(2)	479	-38	YARWAY
(1)	370.5	-146.5	YARWAY
(0)	313.5	+313.5	YARWAY

Bases Figure B 3/4 3-1

REACTOR VESSEL WATER LEVEL

1070 061

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation for longer than 24 hours with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design-basis-accident by increasing the blowdown area and eliminating the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable.

(The recirculation pump speeds must be maintained within the requirements of this specification to ensure that the loop selection logic will operate properly in the case of a loss-of-coolant accident.)

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within (50)°F of each other prior to startup of an idle loop. The loop temperature must also be within (50)°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than (100)°F.

3/4.4.2 SAFETY/RELIEF VALVES

(The reactor coolant system code safety valves and) the safety valve function of the safety-relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of (1325) psig in accordance with the ASME Code.

Demonstration of the (code safety valve and) safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

1070 062

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. (These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.)

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. (Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that certain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.)

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the higher limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so high concentrations of chlorides are not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

1070 063

POOR ORIGINAL

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the ( ) site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-31, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to (20) following a postulated (steam line rupture). The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

1070 064

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (4.9) of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases Figures B 3/4.4.6-1 and B 3/4.4.6-2. The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A', B' and C', includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence as well as adjustments for possible errors in the pressure and temperature sensing instruments.

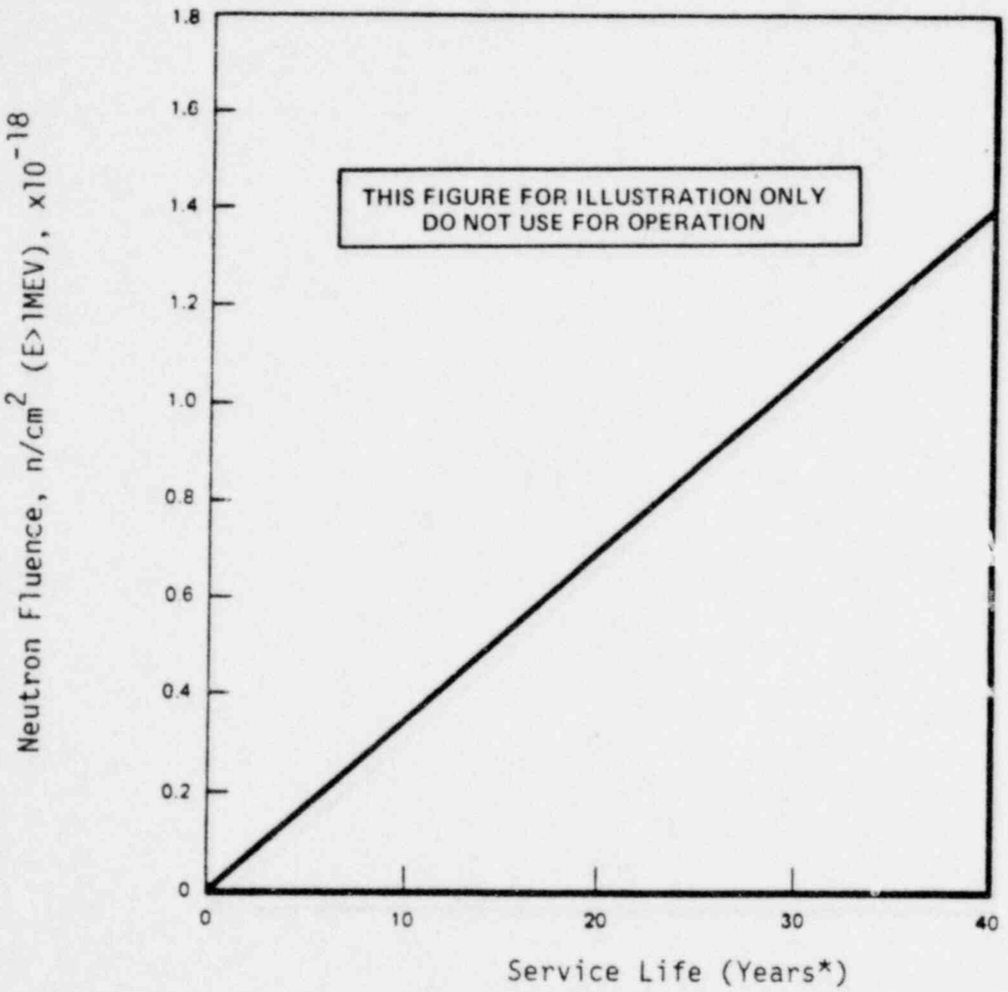
1070 065

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	RT_{NDT} °F	50 FT-LB/35 MIL TEMP F		RT_{NDT} °F	MIN. UPPER SHELF FT-LB	
						LONG	TRANS		LONG	TRANS

1070 066



Fast Neutron Fluence (E>1 Mev) As a Function of Service Life*

Bases Figure B 3/4.4.6-1

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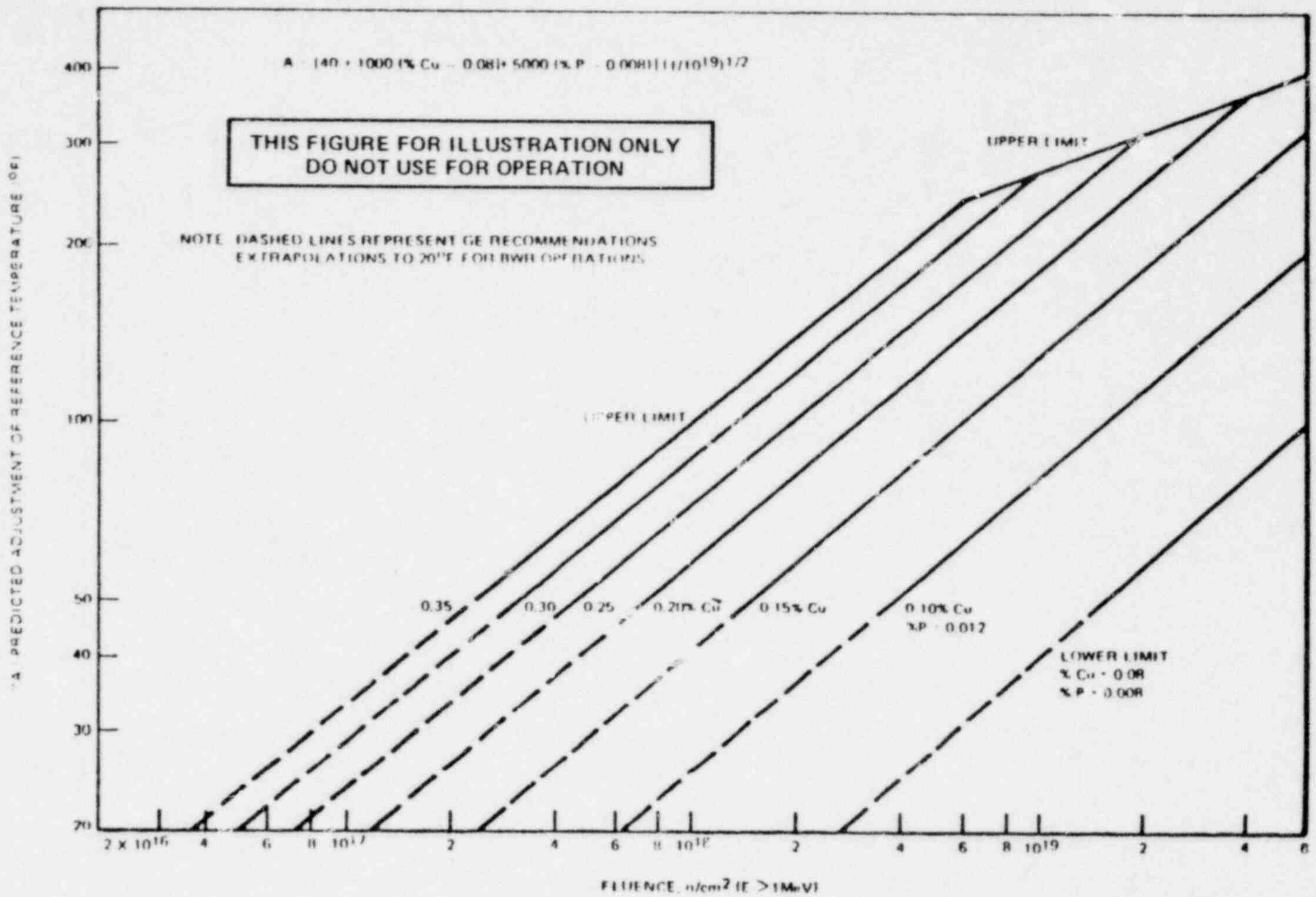
* At (90)% of RATED THERMAL POWER and (90)% availability

1070 067



Effect of Fluence and Copper Content on Shift of RT_{NDT}
For Reactor Vessel Steels Exposed to 550°F Temperature

Bases Figure B 3/4.4.6-2

1070 068

POOR ORIGINAL

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating (in accordance with ASTM E185-73), irradiated flux wires installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wire samples and vessel inside radius are essentially identical, the irradiated flux wires can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire data and Bases Figure B 3/4.4.6-2.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (   ) Edition and Addenda through (   ).

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

1070 069

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1/2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system. ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and to provide a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system.

The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of (1160) psid, differential pressure between reactor vessel and HPCS suction source, to (0) psig.

ECS-OPERATING and SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to (443/1330/4625) gpm at reactor pressures of (1160/1130/200) psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, (but no credit is taken in the hazards analyses for the condensate storage tank water).

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, (a system for which no credit is taken in the hazards analysis), will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of (11) days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds (113) psig even though low pressure core cooling systems provide adequate core cooling up to (350) psig.

ADS automatically controls (seven) selected safety-relief valves although the hazards analysis only takes credit for (six) valves. It is therefore appropriate to permit one valve to be out-of-service for up to (14) days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of (40.4) psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

(The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix "J" of 10 CFR Part 50.)

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the isolation valves when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY
(Reinforced concrete containment)

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of (59) psig in the event of a (LOCA) (steam line break accident). A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

(Prestressed concrete containment with ungrouted tendons.)
This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of (48) psig in the event of a LOCA. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. (The tendon wire or strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.)

(The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," January 1976.)

3/4.6.1.6 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment internal pressure ensure that the containment peak pressure of (40.4) psig does not exceed the design pressure of (62) psig during (LOCA) (steam line break) conditions or that the external pressure does not exceed the design maximum external pressure of (___) psig. The limit of (1.75) psig for initial positive containment pressure will limit the total pressure to (40.4) psig which is less than the design pressure and is consistent with the accident analysis.

3/4.6.1.7 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on primary containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of (340)°F during (LOCA) (steam line break) conditions and is consistent with the accident analysis.

1070 073

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of (59) psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from (1020) psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed (62) psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately (49) psig which is below the design pressure of (62) psig. Maximum water volume of (89,600) ft^3 results in a downcomer submergence of (4'4") and the minimum volume of (87,600 ft^3 results in a submergence approximately (four) inches less. The majority of the Bogeda tests were run with a submerged length of four feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bogeda Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of (90)°F results in a water temperature of approximately (135)°F immediately following blowdown which is below the 170°F used for complete condensation. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

POOR ORIGINAL

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

(In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (see Vermont Yankee letter dated September 13, 1976) which demonstrated a factor of safety of at least (two) for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of (1.7) psid and a suppression chamber water level corresponding to a downcommer submergence range of (4.29) to (4.54) feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.)

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

1070 075

POOR ORIGINAL

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the reactor building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are an adequate number of valves to provide some redundancy so that operation may continue with no more than (3) vacuum breakers inoperable in the closed position.

Each (set of) vacuum breakers between the reactor building and the suppression chamber provides (100)% relief, so operation may continue with up to one (set of) valves out-of-service for (7) days.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is in COLD SHUTDOWN or REFUELING, the drywell may be open and the Reactor Building then becomes the only containment.

Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, latches and dampers, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

1070 076

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. Either hydrogen recombiner (or the primary containment atmosphere dilution system) system is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. (The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.) (The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.)

1070 077

BASES

3/4.7.1 SERVICE WATER SYSTEMS

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

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to (5) rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3. FLOOD PROTECTION (OPTIONAL)

The requirement for flood protection ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions. The limit of elevation (   ) Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds (113) psig even though the LPCI mode of the residual heat removal (RHR) system provides adequate core cooling up to (350) psig.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds (113) psig because RCIC is the primary (non-ECCS) source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified (14) day out-of-service period.

REACTOR CORE ISOLATION COOLING SYSTEM (Continued)

The surveillance requirements provide adequate assurance that RCICS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

3/4.7.5 HYDRAULIC SNUBBERS

The hydraulic 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. The only snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety related system.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment (only ethylene propylene compounds to date) is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during an inspection of these snubbers determines the time interval for the next required inspection of these snubbers. 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.

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in areas which have high radiation field during shutdown or in especially difficult to remove locations, as identified in Table 3.7.5-1, may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

1070 079

3/4.7.6 SEALED SOURCE CONTAMINATION

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

3/4 7.7 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, CO_2, and Halon systems 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. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a V.L. or F.M. approved method.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. 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.

1070 080

3/4.7.8 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations 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 fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

The barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier-penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when the barriers are not functional, either, 1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or 2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and a hourly fire watch patrol established until the barrier is restored to functional status.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that Safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for an instrument error of (   )°F.

1070 081

BASES

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

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

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

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

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulation Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977.

The surveillance requirement for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

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

1070 082

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES (Continued)

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

The OPERABILITY of the motor operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. (The surveillance requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.1[illegible] "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.)

1070 083

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement 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.

3/4.9.5 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 movement of fuel within the reactor pressure vessel.

1070 084

3/4.9.6 REFUELING PLATFORM OPERABILITY

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel element over irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove (99)% of the assumed (10)% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 REACTOR COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 212°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

1070 085

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.5.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.5 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one LPCI subsystem aligned in the shutdown cooling made in order to minimize contaminated water discharge to the radioactive waste disposal system.

1070 086

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined prestressed concrete structure consisting of a drywell and suppression chamber. The drywell is in the form of a truncated cone on top of a cylindrical suppression chamber attached to the suppression chamber through a series of downcomer vents. The primary containment has a minimum free air volume of (273,000) cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure (45) psig.

b. Maximum internal temperature: drywell (340)°F.
suppression chamber (275)°F.

c. Maximum external pressure (2) psig.

e. Maximum floor differential pressure: (25) psid, downward.
(9) psid, upward.

1070 088

This figure shall consist of a map of the site area and provide at a minimum, the information described in Section (2.1.2) of the FSAR and meteorolgical tower location.

EXCLUSION AREA

FIGURE 5.1.1-1

1070 089

This figure shall consist of a map of the site area showing the Low Population Zone boundary. Features such as towns, roads and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.

LOW POPULATION ZONE

FIGURE 5.1.2-1

1070 090

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain (560) fuel assemblies with each fuel assembly containing (63) fuel rods and (one) water rod clad with (Zircaloy -2). Each fuel rod shall have a nominal active fuel length of (146) inches. The initial core loading shall have a maximum average enrichment of (1.90) weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading (and shall have a maximum average enrichment of (   ) weight percent U-235).

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain (137) control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing (143) inches of Boron Carbide, B4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section (5.2) of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of (1250) psig, and

c. For a temperature of (575)°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately (16,000) cubic feet at a nominal T_{ave} of (540)°F.

1070 091

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of (2.6)% delta k/k for uncertainties as described in Section (4.3) of the FSAR.

b. A nominal (21) inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed (0.98) when aqueous foam moderation is assumed.

DRINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation (603' 4").

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than (1120) fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

1070 092

TABLE 5.7.1-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor	(120) heatup and cooldown cycles	(70)°F to (560)°F to (70)°F
	(12,400) power change cycles	(75)% to (100)% to (75)% of RATED THERMAL POWER
	(80) step change cycles	Loss of Feedwater heaters
	(200) reactor trip cycles	(100)% to (0)% of RATED THERMAL POWER
	(2000) power change cycles	(50)% to (100)% to 50% of RATED THERMAL POWER
	(400) control rod pattern exchanges	___________

SECTION 6.0

ADMINISTRATIVE CONTRO'S

1070 094

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1. The (Plant Superintendent) shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The Unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be onsite when fuel is in the reactor.

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A site Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include (3) members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

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* Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

1070 095

This figure shall show the organizational structure and lines of responsibility for the offsite groups that provide technical and management support for the unit. The organizational arrangement for performance and monitoring Quality Assurance activities should also be indicated.

Figure 6.2.1-1

OFFSITE ORGANIZATION

1070 096

This figure shall show the organizational structure and lines of responsibility for the unit staff. Positions to be staffed by licensed personnel should be indicated.

Figure 6.2.2-1

UNIT ORGANIZATION

1070 097

TABLE 6.2.2-1 (OPTIONAL)

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2 & 3	4 & 5
SOL	1	1*
OL	2	1
Non-Licensed	2	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

#Shift crew position may be less than the minimum requirements for a period of time not to exceed 2 hours i order to accommodate unexpected absense of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1.

1070 098

TABLE 6.2 21 (OPTIONAL)

MINIMUM SHIFT CREW COMPOSITION#

OPERATIONAL CONDITION of Unit 1 - Unit 2 in POWER OPERATION, STARTUP or HOT SHUTDOWN

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	2*
OL**	3	2
Non-Licensed	3	3

OPERATIONAL CONDITION of Unit 1 - Unit 2 in COLD SHUTDOWN or REFUELING

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	1*
OL**	2	2
Non-Licensed	3	3

OPERATIONAL CONDITION of Unit 1 - No Fuel in Unit 2

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL	1	1*
OL	2	1
Non-Licensed	2	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

**Assumes each individual is licensed on both units.

#Shift crew composition, including an individual qualified in radiation protection procedures, may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accomodate minimum requirements of Table 6.2.2-1.

1070 099

6.3 UNIT STAFF QUALIFICATIONS

Minimum qualifications for members of the unit staff may be specified by use of an overall qualification statement referencing ANSI N18.1-1971 or alternately by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of a unique organizational structure.

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the (Radiation Protection Manager) who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the (position title) and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.5 REVIEW AND AUDIT

The method by which independent review and audit of facility operations is accomplished may take one of several forms. The licensee may either assign this function to an organizational unit separate and independent from the group having responsibility for unit operation or may utilize a standing committee composed of individuals from within and outside the licensee's organization.

Irrespective of the method used, the licensee shall specify the details of each functional element provided for the independent review and audit process as illustrated in the following example specifications.

6.5.1 UNIT REVIEW GROUP (URG)

FUNCTION

6.5.1.1 The (Unit Review Group) shall function to advise the (Plant Superintendent) on all matters related to nuclear safety.

1070 100

COMPOSITION

6.5.1.2 The (Unit Review Group) shall be composed of the:

Chairman:	(Plant Superintendent)
Member:	(Operations Supervisor)
Member:	(Technical Supervisor)
Member:	(Maintenance Supervisor)
Member:	(Plant Instrument and Control Engineer)
Member:	(Plant Nuclear Engineer)
Member:	(Health Physicist)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the (URG) Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in (URG) activities at any one time.

MEETING FREQUENCY

6.5.1.4 The (URG) shall meet at least once per calendar month and as convened by the (URG) Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the (URG) necessary for the performance of the (URG) responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The (Unit Review Group) shall be responsible for:

a. Review of (1) all procedures required by Specification 6.8 and changes thereto, and (2) any other proposed procedures or changes thereto as determined by the (Plant Superintendent) to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

RESPONSIBILITIES (Continued)

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the (Superintendent of Power Plants) and to the (Company Nuclear Review and Audit Group).

f. Review of events requiring 24 hour written notification to the Commission.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the (Plant Superintendent) or the (Company Nuclear Review and Audit Group).

i. Review of the Security Plan and implementing procedures and shall submit recommended changes to the (Company Nuclear Review and Audit Group).

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the (Company Nuclear Review and Audit Group).

AUTHORITY

6.5.1.7 The (Unit Review Group) shall:

a. Recommend in writing to the (Plant Superintendent) approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the (Superintendent of Power Plants) and the (Company Nuclear Review and Audit Group) of disagreement between the (URG) and the (Plant Superintendent); however, the (Plant Superintendent) shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The (Unit Review Group) shall maintain written minutes of each (URG) meeting that, at a minimum, document the results of all (URG) activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the (Superintendent of Power Plants) and the (Company Nuclear Review and Audit Group).

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6.5.2 COMPANY NUCLEAR REVIEW AND AUDIT GROUP (CNRAG)

FUNCTION

6.5.2.1 The (Company Nuclear Review and Audit Group) shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

COMPOSITION

6.5.2.2 The (CNRAG) shall be composed of the:

Director:	(Position Title)
Member:	(Position Title)
Member:	(Position Title)
Member:	(Position Title)
Member:	(Position Title)

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the (CNRAG) Director to serve on a temporary basis; however, no more than two alternates shall participate as voting members in (CNRAG) activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the (CNRAG) Director to provide expert advice to the (CNRAG).

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MEETING FREQUENCY

6.5.2.5 The (CNRAG) shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 The minimum quorum of the (CNRAG) necessary for the performance of the (CNRAG) review and audit functions of these Technical Specifications shall consist of the Director or his designated alternate and (at least 4 CNRAG) members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The (CNRAG) shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the (Unit Review Group).

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the (CNRAG). These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Appendix A Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

g. Any other area of unit operation considered appropriate by the (CNRAG) or the (Vice President Operations).

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

AUTHORITY

6.5.2.9 The (CNRAG) shall report to and advise the (Vice President Operations) on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

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RECORDS

6.5.2.10 Records of (CNRAG) activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each (CNRAG) meeting shall be prepared, approved and forwarded to the (Vice President-Operations) within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the (Vice President-Operations) within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the (Vice President-Operations) and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the (URG) and submitted to the (CNRAG) and the (Superintendent of Power Plants).

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within two hours.

b. The Safety Limit violation shall be reported to the Commission, the (Superintendent of Power Plants) and to the (CNRAG) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the (URG). This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the (CNRAG) and the (Superintendent of Power Plants) within 14 days of the violation.

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6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the (URG) and approved by the (Plant Superintendent) prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the (URG) and approved by the (Plant Superintendent) within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement, unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

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STARTUP REPORT (Continued)

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation, supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. (Any other unit unique reports required on an annual basis).

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require unit shutdown, operation of safety systems, or other protective measures required by technical specifications.

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PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

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SPECIAL REPORTS

Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

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

6.10 RECORD RETENTION

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

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

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

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

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

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

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

f. Records of radioactive shipments.

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

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

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

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

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

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

RECORD RETENTION (Continued)

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

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

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

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

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

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

k. Records of meetings of the (URG) and the (CNRAG).

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA (OPTIONAL)

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each 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 requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (OPTIONAL) (Continued)

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the unit Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or the unit Health Physicist.

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NRC FORM 335 (7-77)

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The Standard Technical Specifications for General Electric Boiling Water Reactors (GE-STS) is a generic document prepared by the USNRC for use in the licensing process of current General Electric Boiling Water Reactors. The GE-STS sets forth the Limit, Operating Conditions and other requirements applicable to nuclear reactor facility operation as set forth by Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public. This document is revised periodically to reflect current licensing requirements.

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