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

JUL 25 1985

Docket No. 50-458

Mr. William J. Cahill, Jr.
Senior Vice President
River Bend Nuclear Group
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704
ATTN: Mr. J. E. Booker

Dear Mr. Cahill:

SUBJECT: TECHNICAL SPECIFICATIONS FOR RIVER BEND STATION UNIT 1

Enclosed is a final draft of the Technical Specifications for River Bend Station Unit 1. Please review the enclosed and submit, in a timely manner to support license issuance, a certification under oath or affirmation that, to the best of your knowledge, the enclosed draft accurately reflects the plant FSAR, the staff's SER, and the as-built configuration of the plant. In addition, we request that you describe in your response, in some detail, the process and resources used to accomplish this review.

The Technical Specifications to be issued as Appendix A to the River Bend Station Unit 1 license are expected to be identical to the enclosed final draft except for four potential areas still being resolved which are delineated in Enclosure 2 to this letter. GSU should review these Technical Specifications and identify any changes, including the items identified in Enclosure 2. Any changes you wish to make, other than those required to correct editorial and/or typographical errors, must be docketed with adequate justification for the requested changes.

While NRC recognizes that Technical Specifications change over the life of the facility, the Technical Specifications at the time of licensing should be adequate for full power operation through the first refueling outage.

At a recent meeting between J. Deddens of GSU and H. Thompson of NRR, GSU did discuss the possibility of amending the Technical Specifications at a later date.

At this time we are requesting that you provide a summary discussion of the long term changes you are currently considering for the Technical Specifications. Should you have any questions regarding this matter, please contact the NRC Project Manager, Stephen Stern at (301) 492-8348.

Sincerely,

T M Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosures:
As stated

cc: See next page

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Enclosure 2

Technical Specifications in Process of Resolution Between NRC Staff and GSU.

1. Temporary use of unqualified valves for drywell vent and purge (TS 3/4.6.2.7)
GSU needs to purge containment of ammonia (outgassing of insulation) for 24 hours prior to exceeding 5% power. GSU requests use of unqualified valves in the drywell purge and vent system for this purpose, which the staff finds unacceptable.

2. Valve groupings (TS 3/4.3.2 and 3/4.6.4)
GSU submitted letters to NRC dated July 19, 1985 proposing to modify the Technical Specifications concerning valve groupings in primary containment and drywell isolation valves.

3. Periodic testing of the SLCS interlock with valve C41-F031
A potential deficiency in the Technical Specifications omits testing of the SLCS interlock with valve C41-F031 as specified in the SER Section 7.4.2.3.

4. Periodic calibration of thermal overload switches for overcurrent protection of electrical penetrations
A potential deficiency in the Technical Specifications omits periodic testing of the overcurrent devices, as specified in SER section 8.4.2.

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TECHNICAL SPECIFICATIONS

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SECTION 1.0

DEFINITIONS

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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 that part of a Specification which prescribes remedial measures required under designated conditions.

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.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

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. Normal movement of the SRMs, IRMs, TIPs or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be the highest value of the FLPD which exists in the core.

CRITICAL POWER RATIO

1.9 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

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per 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."

DRYWELL INTEGRITY

1.11 DRYWELL INTEGRITY shall exist when:

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

   1. Capable of being closed by an OPERABLE drywell automatic isolation 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 Specification 3.6.4.

b. All drywell equipment hatches are closed and sealed.

c. The drywell airlock is in compliance with the requirements of Specification 3.6.2.3.

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d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ 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

1.13 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. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.15 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.16 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

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FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM

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

IDENTIFIED LEAKAGE

1.19 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the drywell 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

1.20 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. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.21 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

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat 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

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc,

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of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MEMBER(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.26 The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints. It shall also contain a table and figure defining current radiological environmental monitoring sample locations.

OPERABLE - OPERABILITY

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

OPERATIONAL CONDITION - CONDITION

1.28 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

1.29 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.

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PRESSURE BOUNDARY LEAKAGE

1.30 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING

1.31 PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING shall exist when:

a. All containment penetrations required to be closed during accident conditions are closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.

b. All containment hatches are closed.

c. Each containment air lock is in compliance with the requirements of Specification 3.6.1.4.

PRIMARY CONTAINMENT INTEGRITY - OPERATING

1.32 PRIMARY CONTAINMENT INTEGRITY - OPERATING shall exist when:

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

   1. Capable of being closed by an OPERABLE containment automatic isolation 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 Specification 3.6.4.

b. All containment equipment hatches are closed and sealed.

c. Each containment air lock is in compliance with the requirements of Specification 3.6.1.4.

d. The containment leakage rates are within the limits of Specification 3.6.1.3.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

1.33 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71 and

Federal and State regulations and other requirements governing the disposal of the radioactive waste.

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2894 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 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-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.36 A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.

ROD DENSITY

1.37 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 - FUEL BUILDING

1.38 SECONDARY CONTAINMENT INTEGRITY - FUEL BUILDING shall exist when:

a. All Fuel Building penetrations required to be closed during accident conditions are closed by valves, blind flanges, or dampers secured in position.

b. All Fuel Building equipment hatch covers are installed.

c. The Fuel Building Charcoal Filtration System is in compliance with the requirements of Specification 3/4.6.5.6.

d. At least one door in each access to the Fuel Building is closed, except for routine entry and exit of personnel and equipment.

e. The pressure within the Fuel Building is maintained in compliance with the requirements of Specification 4.6.5.1.a.

SECONDARY CONTAINMENT INTEGRITY - OPERATING

1.39 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall exist when:

a. All Auxiliary Building penetrations, Fuel Building penetrations and Shield Building annulus penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation signal, or

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve or damper, as applicable, secured in its closed position, except as provided in Specification 3.6.5.2.

b. All Auxiliary Building, Fuel Building and Shield Building annulus equipment hatches are closed and sealed.

c. The Standby Gas Treatment System is in compliance with the requirements of Specification 3.6.5.4.

d. The Fuel Building Charcoal Filtration System is in compliance with the requirements of Specification 3.6.5.6.

e. At least one door in each access to the Auxiliary Building, Fuel Building and Shield Building annulus is closed, except for routine entry and exit of personnel and equipment.

f. The sealing mechanism associated with each Auxiliary Building, Fuel Building and Shield Building annulus penetration, e.g., welds, bellows or O-rings, is OPERABLE.

g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.40 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.

SITE BOUNDARY

1.41 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

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

SOURCE CHECK

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

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STAGGERED TEST BASIS

1.44 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 equal subintervals.

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

1.46 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components: (a) the time from initial movement of the main turbine stop valve or control valve until 80% of turbine bypass capacity is established, and (b) the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve. The response times may be measured by any series of sequential, overlapping or total steps such that both entire response time components are measured.

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

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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.
P	Prior to each radioactive release.
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#,***	> 200°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 200°F
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°F

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

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

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled provided that the one-rod-out interlock is OPERABLE.

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SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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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 and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06 and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

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 above 1325 psig, as measured in the reactor vessel steam dome, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

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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 ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

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	$\leq$ 120/125 divisions of full scale	$\leq$ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	$\leq$ 0.66 W+48%, with a maximum of	$\leq$ 0.66 W+51%, with a maximum of
2) High Flow Clamped	$\leq$ 111.0% of RATED THERMAL POWER	$\leq$ 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1064.7 psig	$\leq$ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	$\geq$ 9.7 inches above instrument zero*	$\geq$ 8.7 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	$\leq$ 51.0 inches above instrument zero*	$\leq$ 52.1 inches above instrument zero
6. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 12% closed
7. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
8. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
9. Scram Discharge Volume Water Level - High		
a. Level Transmitter - LISN601A and B	$\leq$ 49"	$\leq$ 53"
LISN601C and D	$\leq$ 49"	$\leq$ 51.7"
b. Float Switches - LSN013A and B	$\leq$ 47.32"	$\leq$ 53.50"
LSN013C and D	$\leq$ 45.44"	$\leq$ 49.00"

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

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TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 530 psig	$\geq$ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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2.0 INTRODUCTION

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 1.06. MCPR greater than 1.06 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 28,000 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 28,000 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.

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2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no 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.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2.

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[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 during any fuel cycle would not be as severe as the distribution used in the analysis.

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

b. General Electric "Process Computer Performance Evaluation Accuracy" NEDO-20340 and Amendment 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

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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 1971 Edition, including Addenda through Summer 1973, 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 pressure Safety Limit is selected to be the lowest transient overpressure allowed by the ASME Boiler and Pressure Vessel Code, Section III, Class I.

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shut down, 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 exceeding 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, 2 IRM channels associated with each of the reactor protection system trip channels. 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, and backup to the APRM prior to entering the Run mode.

For BWR 6 plants, the role of the IRM system in responding to potential Rod Withdrawal Error (RWE) accidents is greatly diminished due to the use of a dual channel Rod Pattern Control System.

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 RPCS. Of all the possible sources of reactivity input, uniform control rod 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.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

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-High 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-High setpoint, a time constant of 6 ± 0.6 seconds 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 CMFLPD is ≥ to FRTP.

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 control valve fast closure and turbine stop valve closure trips are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

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 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.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

5. Reactor Vessel Water Level-High

A reactor scram from high reactor water level, approximately two feet above normal operating level, is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. An excess of feedwater entering the vessel would be detected by the level increase in a timely manner. This scram feature is only effective when the reactor mode switch is in the Run position because at THERMAL POWER levels below 10% to 15% of RATED THERMAL POWER, the approximate range of power level for changing to the Run position, the safety margins are more than adequate without a reactor scram.

6. 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.

7. 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.

8. Drywell Pressure-High

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

9. 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, control rod insertion would be hindered. The reactor is therefore tripped when

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Scram Discharge Volume Water Level-High (Continued)

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 when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of approximately 17 gallons of water.

10. 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 5% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient.

11. 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 coincident failure of the turbine bypass valves. 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 20 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 slower 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. Relevant transient analyses are discussed in Section 15.2.2 of the Final Safety Analysis Report.

12. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position provides additional manual reactor trip capability.

13. Manual Scram

The manual scram pushbutton switches provide a diverse means for initiating a reactor shutdown (scram) to the automatic protective instrumentation channels and provide manual reactor trip capability.

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SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

This Specification is not applicable in OPERATIONAL CONDITIONS 4 or 5.

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

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4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

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

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

b. The combined time interval for any 3 consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

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

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable condition 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. 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:

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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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 insertable control rods to be inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL CONDITION 4, establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING within 8 hours.

c. In OPERATIONAL CONDITION 5, suspend CORE ALTERATIONS and other activities that could reduce the SHUTDOWN MARGIN and insert all insertable control rods within 1 hour. Establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.1 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 specified limit.

c. Within 12 hours 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.

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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 equivalence of the difference exceeding 1% delta k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

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. Within one hour:

        a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

        b) Disarm the associated directional control valves* either:

            1) Electrically, or

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

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

    2. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

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

    1. If the inoperable control rod(s) is withdrawn, within one hour:

        a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

        b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range**.

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

**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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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves* either:

a) Electrically, or

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

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves* either:

a) Electrically, or

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

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

3. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open,** and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RPCS, 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, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

**These valves may be closed intermittently for testing under administrative controls.

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SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.3, 4.1.3.4 and 4.1.3.5.

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

   1. Close within 30 seconds after receipt of a signal for control rods to scram, and

   2. Open when the scram signal is reset.

b. Proper level sensor response by performance, at least once per 31 days, of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation.

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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, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.31	0.81	1.44
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Surveillance Requirement 4.1.3.2.a or b, operation may continue provided that:

   1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

   2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Average Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

   3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 5.

*For intermediate reactor vessel dome pressure, the scram time criterion is determined by linear interpolation at each notch position.

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ACTION: (Continued)

4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupies an adjacent location in any direction, including the diagonal, to another such control rod.

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

b. With a "slow" control rod(s) not satisfying ACTION a.1, above:

1. Declare the "slow" control rod(s) inoperable, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

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

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.

2. Each of these "slow" control rods satisfies the limits of ACTION a.1.

3. The eight adjacent control rods surrounding each "slow" control rod are:

   a) Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2, and

   b) OPERABLE.

4. The total number of "slow" control rods, as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3, as determined by Specification 4.1.3.2.a and b, does not exceed 5.

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

d. The provisions of Specification 3.0.4 are not applicable.

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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 at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided this surveillance is completed prior to entry into OPERATIONAL CONDITION 1.

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CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.3 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, within 8 hours:

      a) Restore the inoperable accumulator to OPERABLE status, or

      b) Declare the control rod associated with the inoperable accumulator inoperable.

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

   2. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

      a) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch or place the reactor mode switch in the Shutdown position.

      b) Insert the inoperable control rods and disarm the associated directional control valves either:

         1) Electrically, or

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

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

b. In OPERATIONAL CONDITION 5*:

   1. With one withdrawn control rod with its associated scram accumulator inoperable, insert the affected control rod and disarm the associated directional control valves within one hour, either:

*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.

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LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

a) Electrically, or

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

2. With more than one withdrawn control rod with the associated scram accumulator inoperable and with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 1520 psig unless the control rod is inserted and disarmed or scrammed.

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, and verifying an alarm setpoint of $\geq$ 1520 psig on decreasing pressure.

2. Measuring and recording the time for up to 10 minutes that each individual accumulator check valve maintains the associated accumulator pressure above the alarm set point with no control rod drive pump operating.

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CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

3.1.3.4 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, within 2 hours:

   1. If permitted by the RPCS, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

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

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

   2. If recoupling is not accomplished on the first attempt or, if not permitted by the RPCS, then until permitted by the RPCS, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

      a) Electrically, or

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

   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

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

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

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LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. If recoupling is not accomplished, insert the control rod and disarm the associated directional control valves* either:

    a) Electrically, or

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each affected 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 drive 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.

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

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CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2 with one or more control rod position indicators inoperable, within one hour:

  1. Determine the position of the control rod by an alternate control rod position indicator, or

  2. Move the control rod to a position with an OPERABLE position indicator, or

  3. When THERMAL POWER is:

    a) Within the low power setpoint of the RPCS:

      1) Declare the control rod inoperable, and

      2) Verify the position and bypassing of control rods with inoperable "Full-in" and/or "Full-out" position indicators by a second licensed operator or other technically qualified members of the unit technical staff.

    b) Greater than the low power setpoint of the RPCS, declare the control rod inoperable, insert the control rod and disarm the associated directional control valves** either:

      1) Electrically, or

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

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

b. In OPERATIONAL CONDITION 5* with no position indicators of a withdrawn control rod OPERABLE, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of Specification 3.0.4 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.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

4.1.3.5 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, and

b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2.

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CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

3.1.3.6 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

4.1.3.6 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

CONTROL ROD WITHDRAWAL

LIMITING CONDITION FOR OPERATION

3.1.4.1 Control rods shall not be withdrawn.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, when the main turbine bypass valves are not fully closed and THERMAL POWER is greater than the low power setpoint of the rod control and information system (RC & IS).

ACTION:

With any control rod withdrawal when the main turbine bypass valves are not fully closed and THERMAL POWER is greater than the low power setpoint of the RC & IS, immediately return the control rod(s) to the position prior to control rod withdrawal.

SURVEILLANCE REQUIREMENTS

4.1.4.1 Control rod withdrawal shall be prevented, when the main turbine bypass valves are not fully closed and THERMAL POWER is greater than the low power setpoint of the RC & IS, by a second licensed operator or other technically qualified member of the unit technical staff.

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LIMITING CONDITION FOR OPERATION

3.1.4.2 The rod pattern control system (RPCS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#.

ACTION:

a. With the RPCS inoperable or with the requirements of ACTION b, below, not satisfied and with:

   1. THERMAL POWER less than or equal to 20% of RATED THERMAL POWER control rod movement shall not be permitted, except by a scram.

   2. THERMAL POWER greater than 20% of RATED THERMAL POWER control rod withdrawal shall not be permitted.

b. With an inoperable control rod(s), OPERABLE control rod movement may continue by bypassing the inoperable control rod(s) in the RPCS provided that:

   1. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable, this inoperable control rod may be bypassed in the rod gang drive system (RGDS) and/or the rod action control system (RACS) provided that the SHUTDOWN MARGIN has been determined to be equal to or greater than required by Specification 3.1.1.

   2. With up to eight control rods inoperable for causes other than addressed in ACTION b.1, above, these inoperable control rods may be bypassed in the RACS provided that:

      a) The control rod to be bypassed is inserted and the directional control valves are disarmed either:

         1) Electrically, or

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

      b) All inoperable control rods are separated from all other inoperable control rods by at least two control cells in all directions.

      c) There are not more than 3 inoperable control rods in any RPCS group.

---

*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 RPCS prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. The position and bypassing of an inoperable control rod(s) is verified by a second licensed operator or other technically qualified member of the unit technical staff.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RPCS shall be demonstrated OPERABLE by verifying the OPERABILITY of the:

a. Rod pattern controller functions when THERMAL POWER is less than the low power setpoint by selecting and attempting to move an inhibited control rod:

   1. After withdrawal of the first insequence control rod or gang for each reactor startup.

   2. As soon as the rod inhibit mode is automatically initiated at the RPCS low power setpoint, 20 + 15, - 0% of RATED THERMAL POWER, during power reduction.

   3. The first time only that a banked position, N1, N2, or N3, is reached during startup or during power reduction below the RPCS low power setpoint.

b. Rod withdrawal limiter functions when THERMAL POWER is greater than or equal to the low power setpoint by selecting and attempting to move a restricted control rod in excess of the allowable distance:

   1. As each power range above the RPCS low power setpoint is entered during a power increase or decrease.

   2. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

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3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 Two standby liquid control subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

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

   2. With both 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.

b. In OPERATIONAL CONDITION 5*:

   1. With one subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or insert all insertable control rods within the next hour.

   2. With both subsystems inoperable, insert all insertable control rods within one hour.

SURVEILLANCE REQUIREMENTS

4.1.5 Each standby liquid control subsystem 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 Figure 3.1.5-1.

   2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-2 for the percent weight concentration determined once per 31 days per Specification 4.1.5.b.2.

   3. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping up to the first storage tank outlet valve to be greater than or equal to 70°F.

b. At least once per 31 days by;

   1. Verifying the continuity of the explosive charge.

*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 (Continued)

2. Determining, by chemical analysis*, that the available weight of sodium pentaborate is greater than or equal to 4246 lbs and the percent weight concentration of sodium pentaborate in solution is within the limits of Figure 3.1.5-2.

3. 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. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1220 psig is met.

d. 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 all heat traced piping between the storage tank and the reactor vessel is unblocked by:

a) Isolating the pump suction manual maintenance valves and the demineralized water supply line, and

b) Opening each motor-operated pump suction isolation valve independently and verifying flow to the collection shipping drum, and then draining and flushing the piping used for the test with demineralized water after closing both motor-operated pump suction isolation valves.

3. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise for the sodium pentaborate solution in the storage tank after the heaters are energized.

---

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below 70°F.

**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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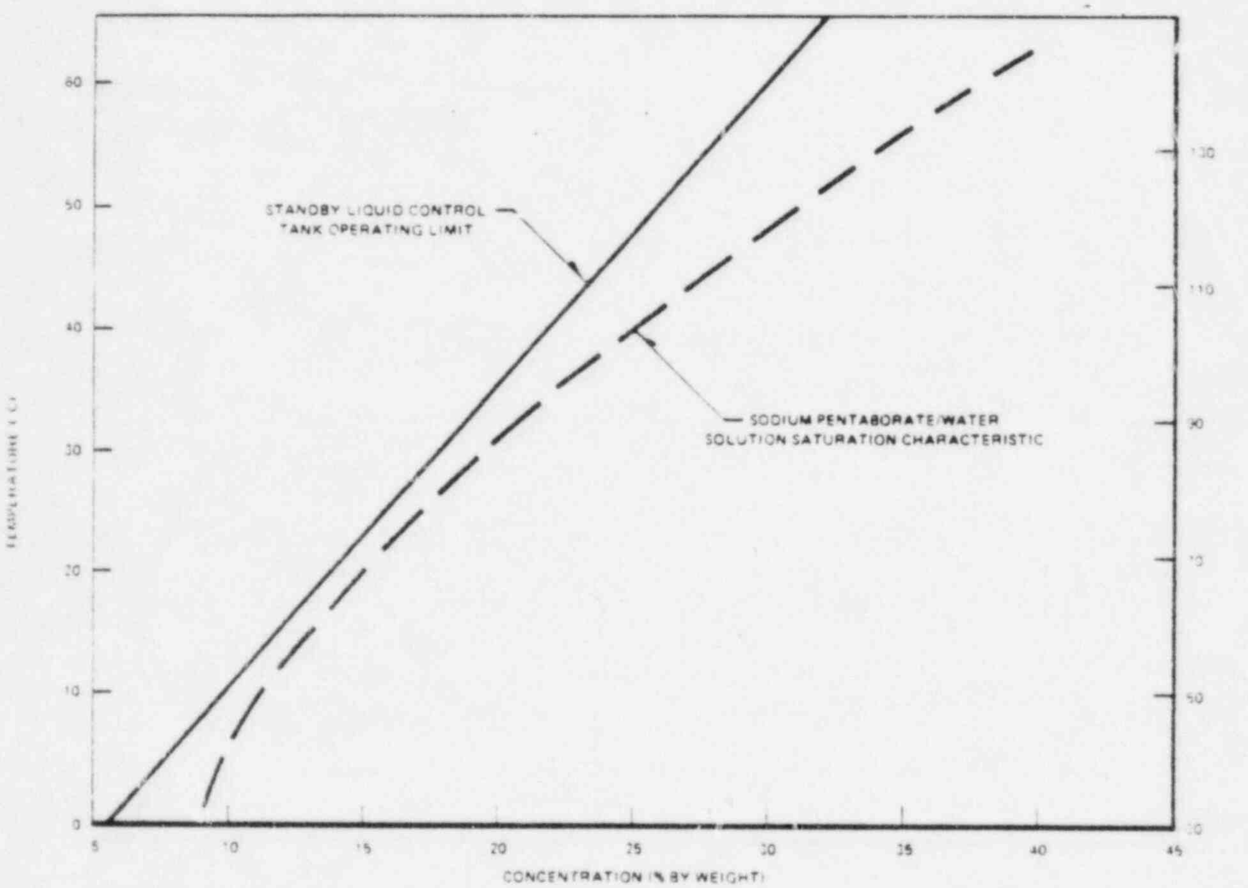

FIGURE 3.1.5-1

SATURATION TEMPERATURE OF SODIUM PENTABORATE SOLUTION

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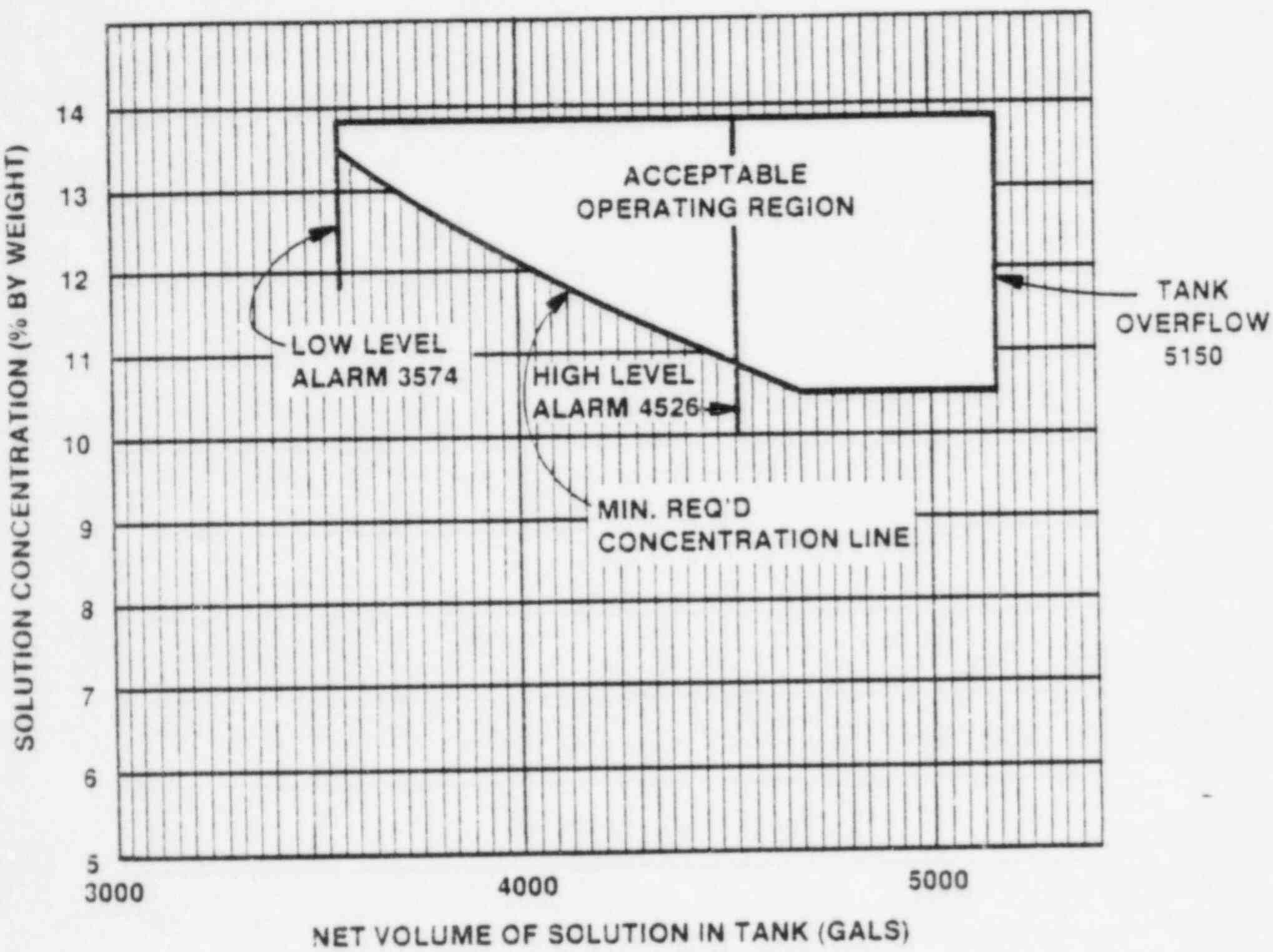

FIGURE 3.1.5-2
SODIUM PENTABORATE SOLUTION VOLUME/CONCENTRATION REQUIREMENTS

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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 Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 or 3.2.1-5, 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 Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5:

a. At least once per 24 hours,

b. Within 12 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.

d. The provisions of Specification 4.0.4 are not applicable.

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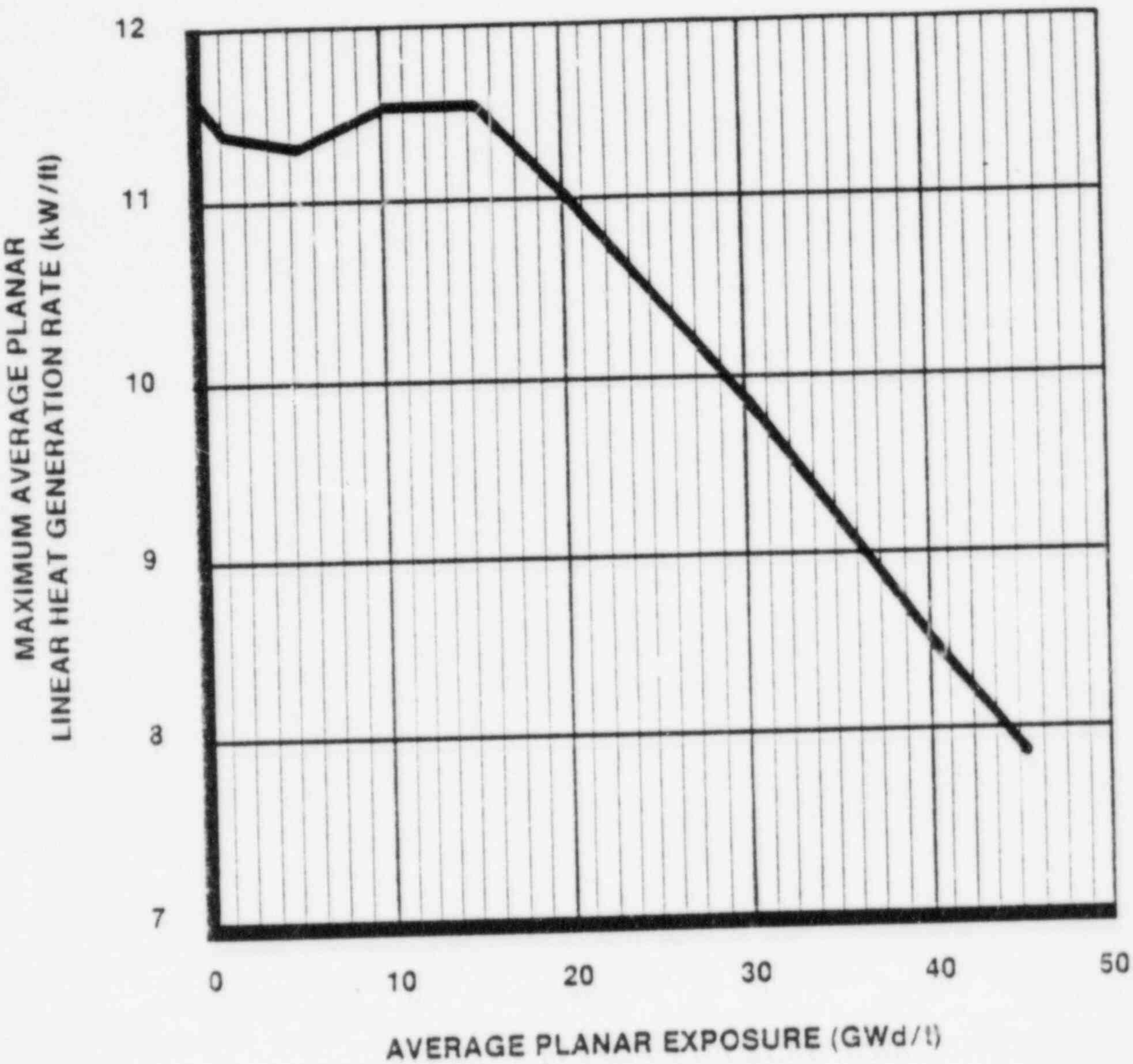

FIGURE 3.2.1-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE-INITIAL CORE FUEL TYPES (P8SIB071)

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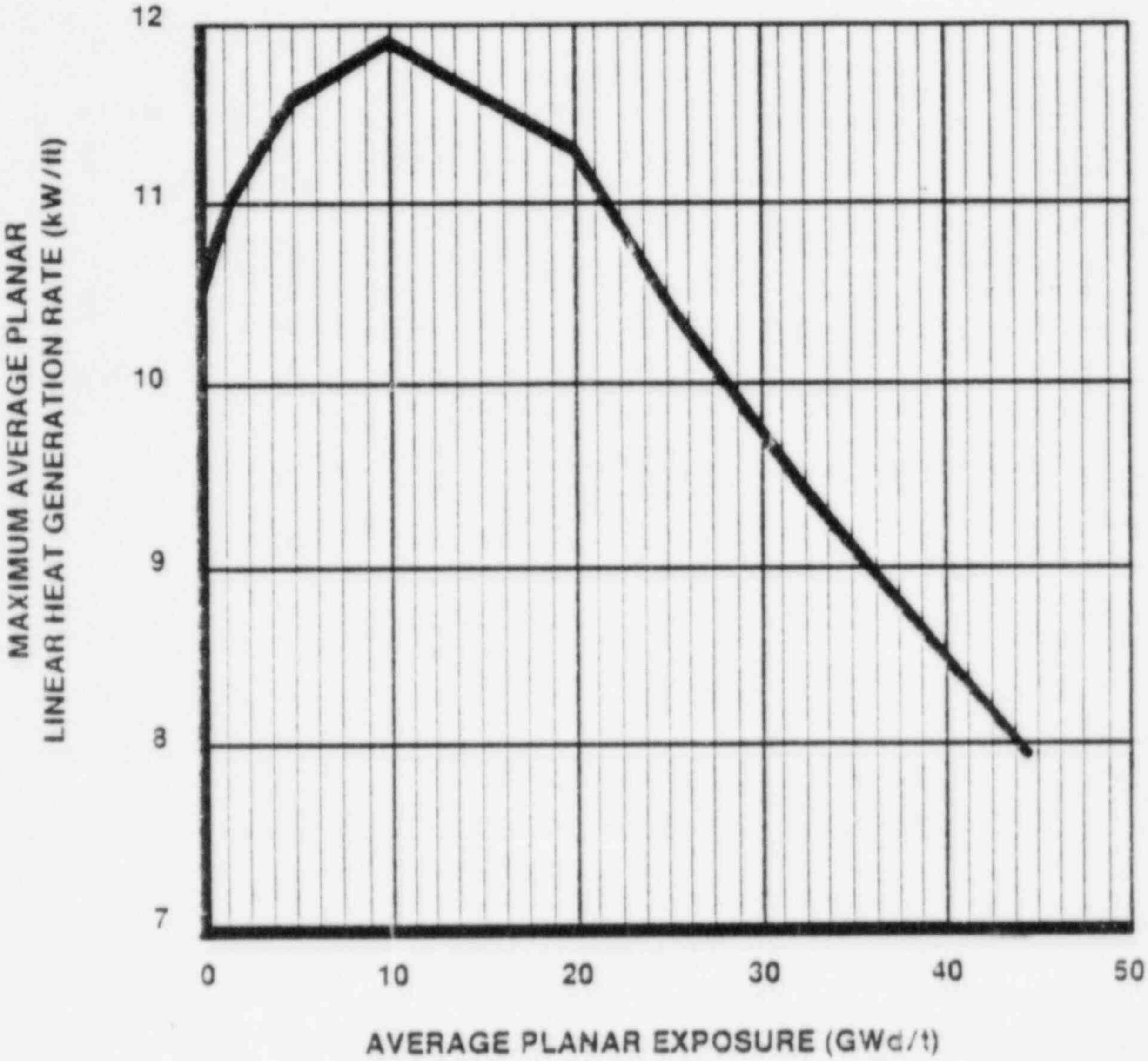

FIGURE 3.2.1-2
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE-INITIAL CORE FUEL TYPES (P8SIB094)

FINAL DRAFT

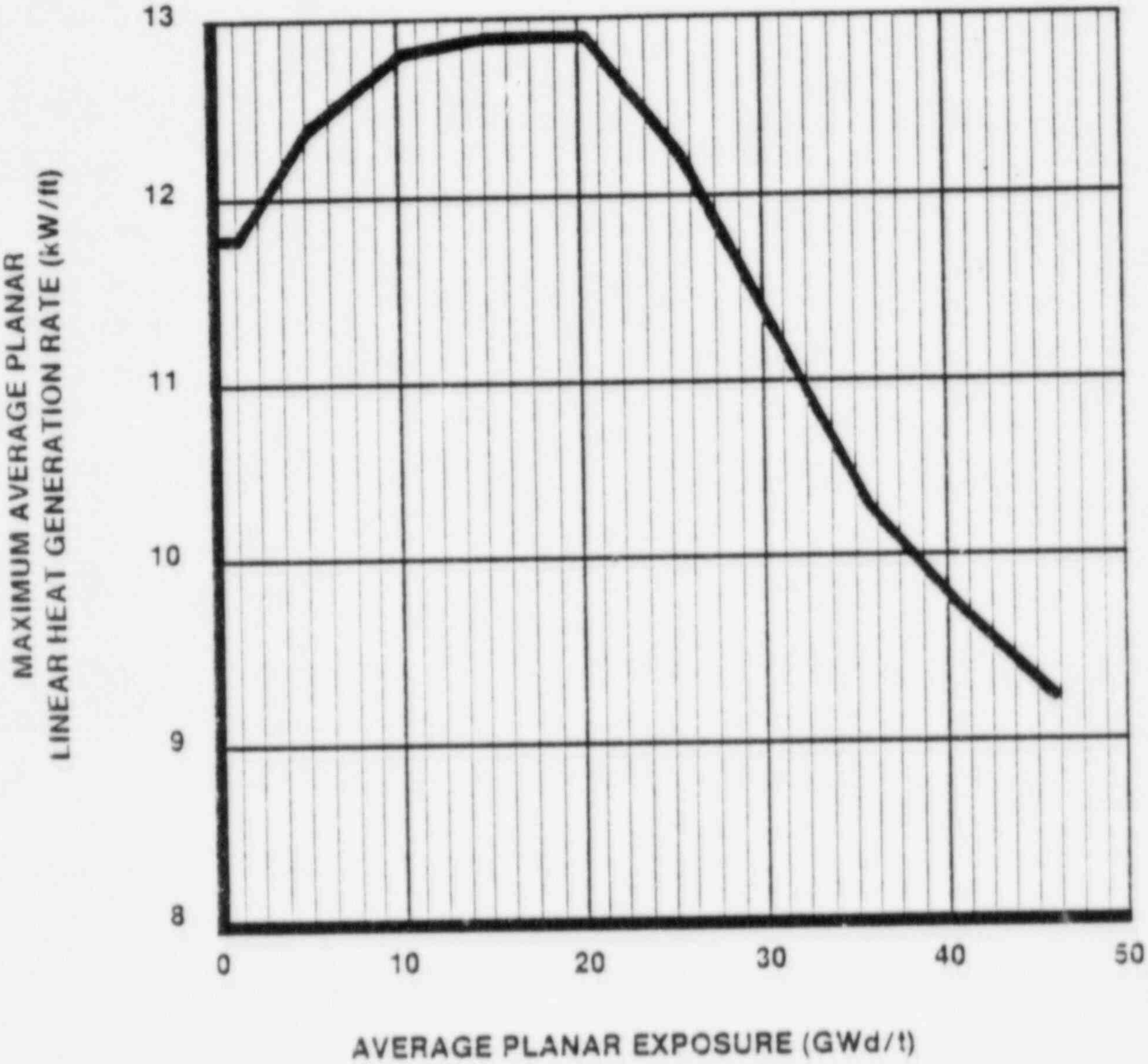

FIGURE 3.2.1-3
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE-INITIAL CORE FUEL TYPES (P8SIB163)

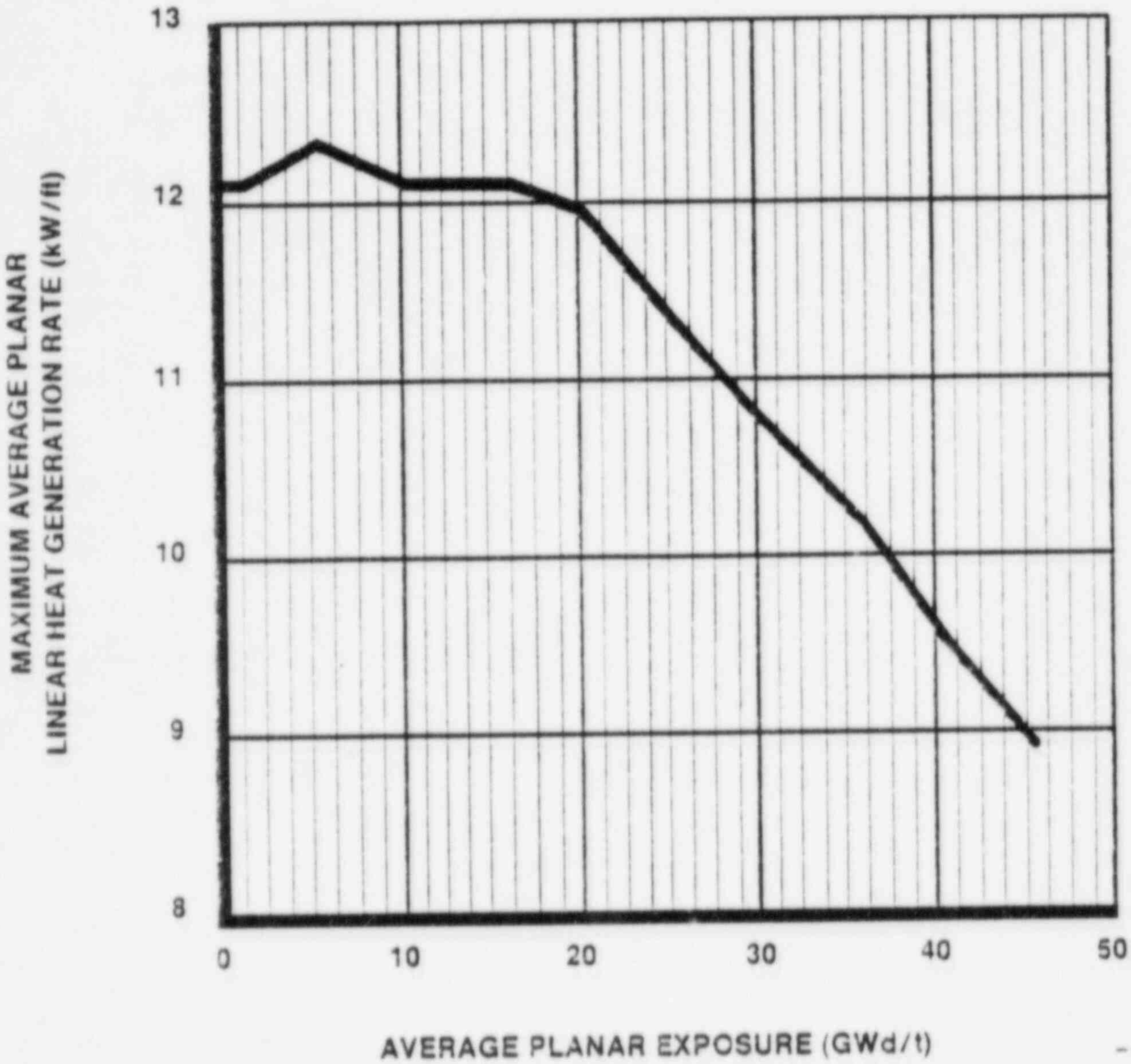

FIGURE 3.2.1-4
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE-INITIAL CORE FUEL TYPES (P8SIB248)

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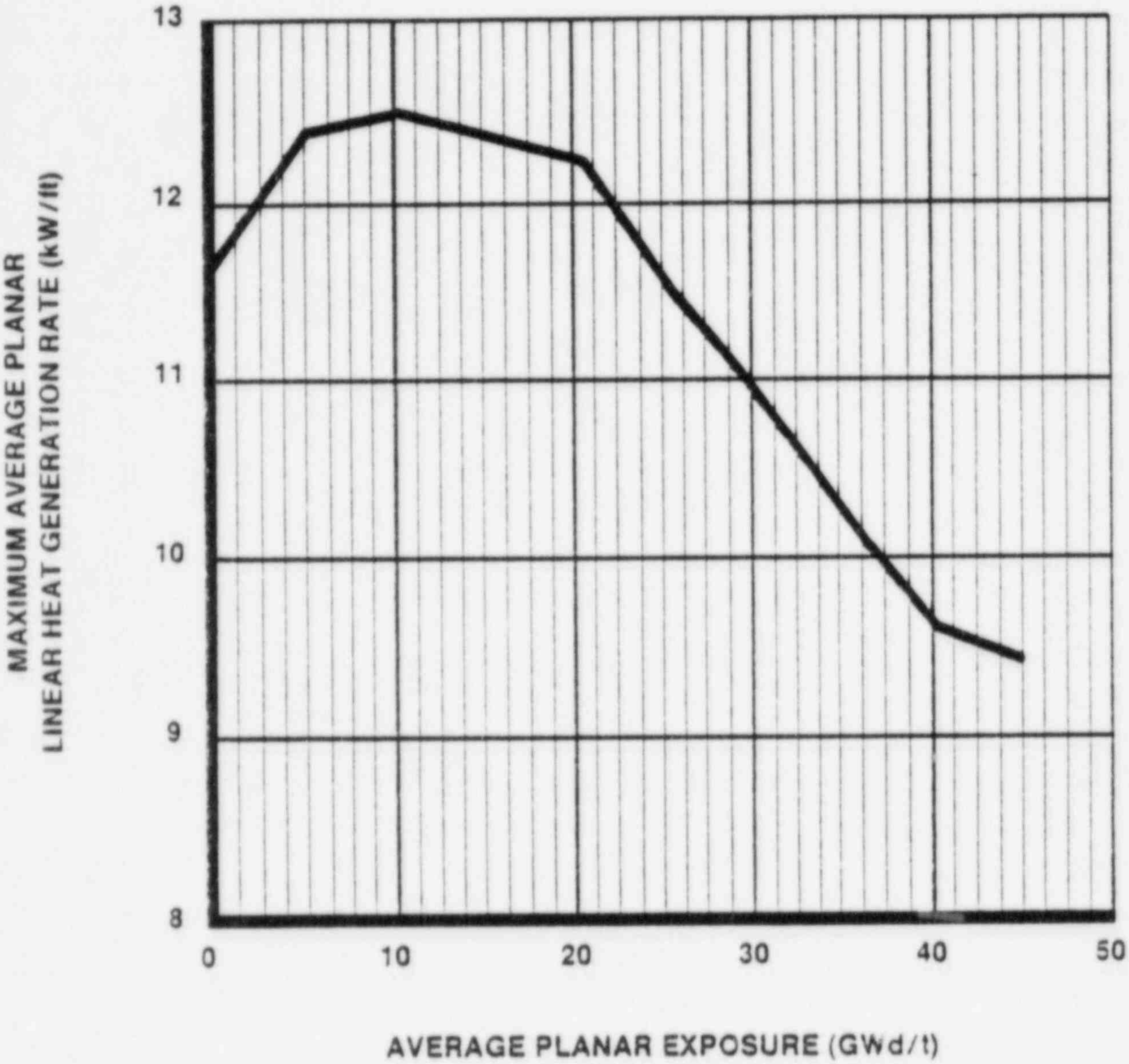

FIGURE 3.2.1-5
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE-INITIAL CORE FUEL TYPES (P8SIB278)

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66W + 48\%)T$	$S \leq (0.66W + 51\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.
T = The ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

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-high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value * within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-high scram and flow biased neutron flux-upscale 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 12 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 CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

---

*With CMFLPD greater than the FRTP, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than both $MCPR_f$ and $MCPR_p$ limits at indicated core flow and THERMAL POWER as shown in Figures 3.2.3-1 and 3.2.3-2.

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

ACTION:

With MCPR less than the applicable MCPR limit shown in Figures 3.2.3-1 and 3.2.3-2, 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 MCPR shall be determined to be equal to or greater than the MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

a. At least once per 24 hours,

b. Within 12 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.

d. The provisions of Specification 4.0.4 are not applicable.

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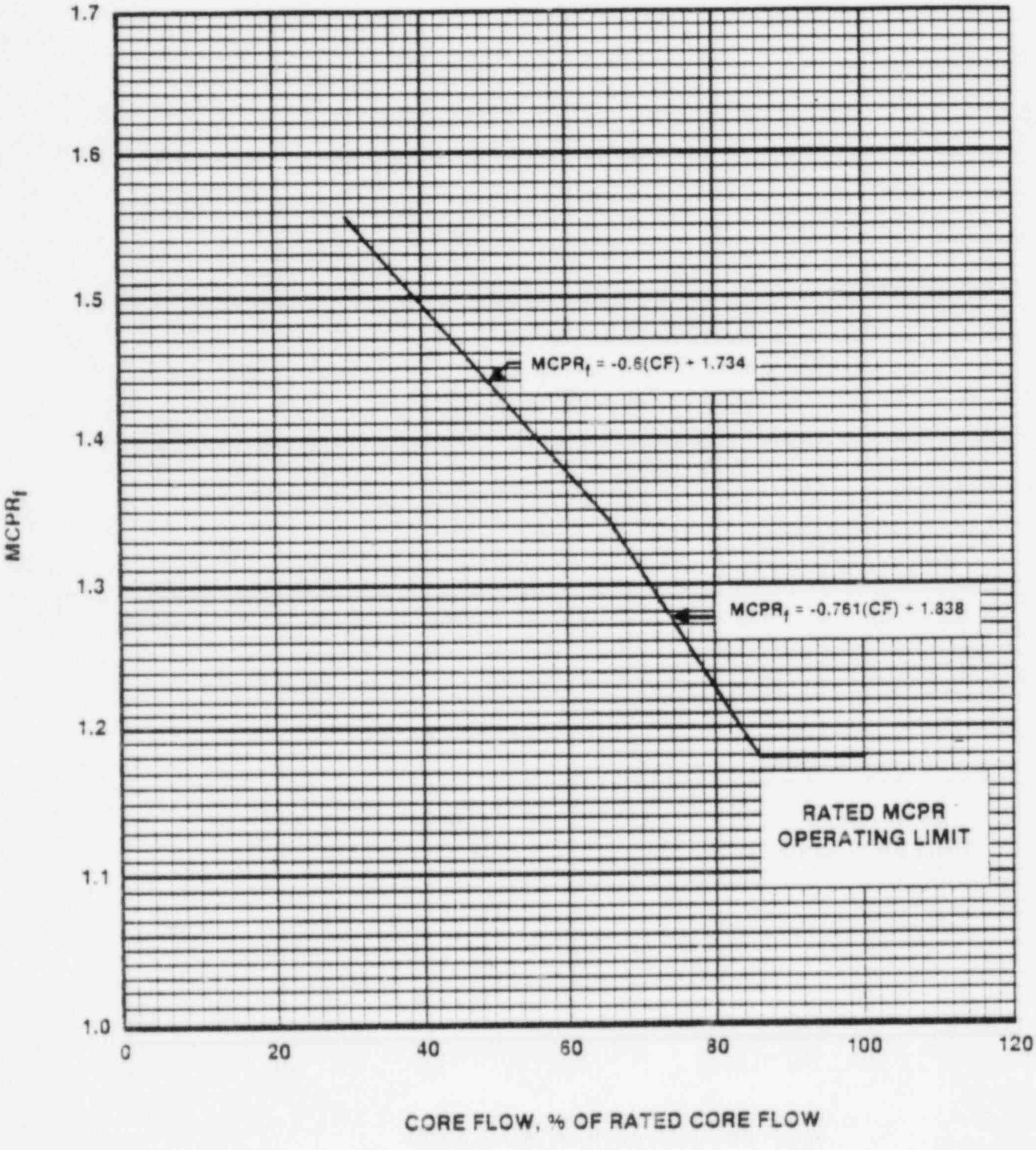

FIGURE 3.2.3-1
MCPR$_f$

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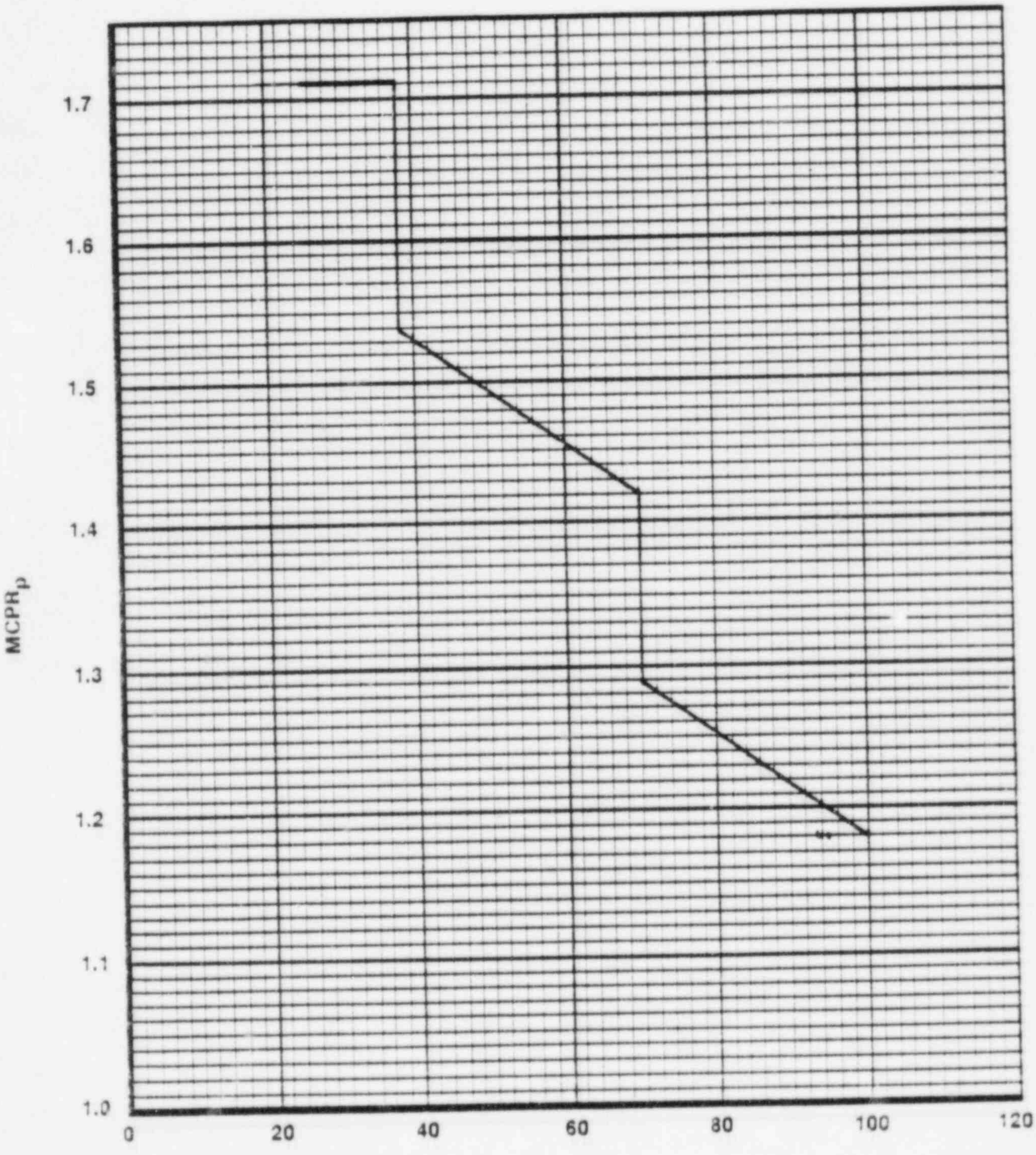

FIGURE 3.2.3-2
$MCPR_p$

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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 than 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 12 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.

d. The provisions of Specification 4.0.4 are not applicable.

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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 the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

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 trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

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 functional unit 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 channel per trip system 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 system.

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*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 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition. The requirement to place a trip system in the tripped condition does not apply to Functional Units 6 and 10 of Table 3.3.1-1.

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 - High	2	3	1
	3, 4	3	2
	$5^{(b)}$	3	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3	3
2. Average Power Range Monitor $^{(c)}$:			
a. Neutron Flux - High, Setdown	2	3	1
	3, 4	3	2
	$5^{(b)}$	3	3
b. Flow Biased Simulated Thermal Power - High	1	3	4
c. Neutron Flux - High	1	3	4
d. Inoperative	1, 2	3	1
	3, 4	3	2
	5	3	3
3. Reactor Vessel Steam Dome Pressure - High	1, $2^{(d)}$	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Reactor Vessel Water Level-High, Level 8	$1^{(e)}$	2	4
6. Main Steam Line Isolation Valve - Closure	$1^{(e)}$	4	10
7. Main Steam Line Radiation - High	1, $2^{(d)}$	2	5
8. Drywell Pressure - High	1, $2^{(f)}$	2	1

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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
9. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	$5^{(g)}$	2	3
b. Float Switches	1, 2	2	1
	$5^{(g)}$	2	3
10. Turbine Stop Valve - Closure	$1^{(h)}$	4	6
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$1^{(h)}$	2	6
12. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
13. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Within 1 hour, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Within 1 hour, place the inoperable instrument channels in both trip systems in the tripped condition; otherwise, initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint (less than 40% of RATED THERMAL POWER) within 2 hours.

ACTION 7 - Within 1 hour, verify all insertable control rods to be inserted.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 10 - Within one hour, place the inoperable instrument channels in both trip systems in the tripped condition; otherwise be in at least STARTUP within 6 hours.

*Except replacement of LPRM strings, provided SRM instrumentation is OPERABLE per Specification 3.9.2.

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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 prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations are being 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) This function is not required to be OPERABLE when the reactor pressure vessel head is 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 DRYWELL 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) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 187 psig,** equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**To allow for instrumentation accuracy, calibration and drift, a setpoint of $\leq$ 177 psig turbine first stage pressure shall be used.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤0.09**
c. Neutron Flux - High	≤0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤0.35
4. Reactor Vessel Water Level - Low, Level 3	≤1.05
5. Reactor Vessel Water Level - High, Level 8	≤1.05
6. Main Steam Line Isolation Valve - Closure	≤0.06
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	
a. Level Transmitter	NA
b. Float Switches	NA
10. Turbine Stop Valve - Closure	≤0.06
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤0.07#
12. Reactor Mode Switch Shutdown Position	NA
13. 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.
**Not including simulated thermal power time constant, 6 ± 0.6 seconds.
#Measured from start of turbine control valve fast closure.

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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 - High	S/U,S,(b)	S/U(c), W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U,S,(b)	S/U(c), W	SA	2
	S	W	SA	3, 4, 5
b. Flow Biased Simulated Thermal Power - High	S,D(h)	S/U(c), W	W(d)(e), SA, R(i)	1
c. Neutron Flux - High	S	S/U(c), W	W(d), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R(g)	1, 2(j)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R(g)	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R(g)	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2(j)
8. Drywell Pressure - High	S	M	R(g)	1, 2(l)

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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 FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R[g]	1, 2, 5[k]
b. Float Switch	NA	Q	R	1, 2, 5[k]
10. Turbine Stop Valve - Closure	S[m]	M[n]	R[g][n]	1
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	S[m]	M[n]	R[g][n]	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

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

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, 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 ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. 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 flow biased channel to conform to a calibrated flow signal.

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TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

(g) Calibrate Rosemount trip unit setpoint at least once per 31 days.

(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.

(i) This calibration shall consist of verifying the simulated thermal power time constant to be less than 6.6 seconds.

(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

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

(l) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required per Specification 3.10.1

(m) Verify the Turbine Bypass Valves are closed when THERMAL POWER is greater than or equal to 40% RATED THERMAL POWER.

(n) The CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

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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 the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

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 trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

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*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.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

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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 4.3.2.1-1.

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

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip 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 channel per trip system 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 trip system.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	1, 7, 8, 9(b)(j), 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 3, 8(b)(c)(j)	2	1, 2, 3	20
c. Containment Purge Isolation Radiation - High	8	1	1, 2, 3	21
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level-Low Low Low, Level 1	6(c)	2	1, 2, 3	20
b. Main Steam Line Radiation - High	6, 9(d)	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	6	2	1	24
d. Main Steam Line Flow - High	6	2/MSL	1, 2, 3	23
e. Condenser Vacuum - Low	6	2	1, 2**, 3**	23
f. Main Steam Line Tunnel Temperature - High	6	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	6	2	1, 2, 3	23
h. Main Steam Line Area Temperature High (Turbine Building)	6	2/area	1, 2, 3	23

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	11, 12, 13(b)(e)(h)(i)	2	1, 2, 3	25
b. Drywell Pressure - High	11, 12, 13(b)(c)(e)(h)(i)	2	1, 2, 3	25
c. Fuel Building Ventilation Exhaust Radiation - High	13(e)(h)	1	*	28
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	12(b)(e)(i)	1	1, 2, 3	29
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	7, 15, 16	1	1, 2, 3	27
b. Δ Flow Timer	7, 15, 16	1	1, 2, 3	27
c. Equipment Area Temperature - High	7, 15, 16	1	1, 2, 3	27
d. Equipment Area Δ Temperature - High	7, 15, 16	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low Level 2	7, 15, 16	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	7, 15, 16	1	1, 2, 3	27

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (continued)				
g. Main Steam Line Tunnel Δ Temperature - High	7, 15, 16	1	1, 2, 3	27
h. SLCS Initiation	7(f), 16	1(f)	1, 2, 3	27
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	2	1	1, 2, 3	27
b. RCIC Steam Line Flow - High Timer	2	1	1, 2, 3	27
c. RCIC Steam Supply Pressure - Low	2	1	1, 2, 3	27
d. RCIC Turbine Exhaust Diaphragm Pressure - High	2	2	1, 2, 3	27
e. RCIC Equipment Room Ambient Temperature - High	2	1	1, 2, 3	27
f. RCIC Equipment Room Δ Temperature - High	2	1	1, 2, 3	27
g. Main Steam Line Tunnel Ambient Temperature - High	2	1	1, 2, 3	27
h. Main Steam Line Tunnel Δ Temperature - High	2	1	1, 2, 3	27

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (continued)				
i. Main Steam Line Tunnel Temperature Timer	2	1	1, 2, 3	27
j. RHR Equipment Room Ambient Temperature - High	2	1	1, 2, 3	27
k. RHR Equipment Room Δ Temperature - High	2	1	1, 2, 3	27
l. RHR/RCIC Steam Line Flow - High	2	1	1, 2, 3	27
m. Drywell Pressure - High	3(g)	1	1, 2, 3	27
n. Manual Initiation	2(k)	1	1, 2, 3	26
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	5, 14	2	1, 2, 3	30
b. RHR Equipment Area Δ Temperature - High	5, 14	2	1, 2, 3	30
c. Reactor Vessel Water Level - Low, Level 3	5, 14	2	1, 2, 3	30
d. Reactor Vessel Water Level - Low Low Low, Level 1	10	2	1, 2, 3	30

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
6. RHR SYSTEM ISOLATION (continued)				
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	5	2	1, 2, 3	30
f. Drywell Pressure - High	10, 14	2	1, 2, 3	30
7. MANUAL INITIATION	1, 5, 6, 7, 8, 10, 11, 12, 13, 14, 15, 16	2	1, 2, 3	22

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TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 22 - 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 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Within 1 hour, establish SECONDARY CONTAINMENT INTEGRITY - OPERATING with the standby gas treatment system and Fuel Building Ventilation System (emergency mode) operating.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 28 - Within 1 hour, initiate and maintain the Fuel Building Ventilation System in the emergency mode of operation.

ACTION 29 - Within 1 hour, initiate and maintain annulus mixing system with the reactor building annulus exhaust to at least one operating standby gas treatment train.

ACTION 30 - Within 1 hour, lock the affected system isolation valves closed and declare the affected system inoperable.

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TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

NOTES

* When handling irradiated fuel in the Fuel Building.
** May be bypassed with reactor mode switch not in Run and all turbine stop valves closed.
*** The valve groups listed are designated in Tables 3.6.4-1 and 3.6.5.3-1.
(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) Also actuates the standby gas treatment system.
(c) Also actuates the main control room air conditioning system in the emergency mode of operation.
(d) Also trips and isolates the air removal pumps.
(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.3-1.
(f) Manual initiation of SLCS pump C001B closes 1G33*MOVF001, and manual initiation of SLCS pump C001A closes 1G33*MOVF004.
(g) Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.
(h) Also starts the Fuel Building Exhaust Filter Trains A and B.
(i) Also starts the Annulus Mixing System.
(j) Also actuates the containment hydrogen analyzer/monitor recorder.
(k) Manual initiation isolates the outboard steam supply isolation valve only and only following a manual or automatic initiation of the RCIC 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 - Low Low Level 2	≥-43 inches*	≥-47 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Containment Purge Isolation Radiation - High	≤ 1.3 R/hr	≤ 1.57 R/hr
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low Level 1	≥-143 inches*	≥-147 inches
b. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 173** psid	≤ 178** psid
e. Condenser Vacuum - Low	≥ 8.5 inches Hg. vacuum	≥ 7.6 inches Hg. vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 135°F	≤ 142.5°F
g. Main Steam Line Tunnel Δ Temperature - High	≤ 51°F	≤ 55°F

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. MAIN STEAM LINE ISOLATION (Cont'd)		
h. Main Steam Line Area Temperature - High (Turbine Building)		
1. Main Steam Tunnel Area (El. 95')	≤ 142°F	≤ 145.3°F
2. Main Steam Tunnel Area (El. 114')	≤ 142°F	≤ 145.3°F
3. Main Steam Line Turbine Shield Wall	≤ 102°F	≤ 106°F
4. MSL Moisture Separator and Reheater Area	≤ 126°F	≤ 130°F
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low Level 2	≥ - 43 inches*	≥ - 47 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Fuel Building Ventilation Exhaust Radiation - High		
1RMS*RE5A	≤ 1.82×10^{3} µCi/sec	≤ 2.18×10^{3} µCi/sec
1RMS*RE5B	≤ 5.88×10^{-4} µCi/cc	≤ 7.05×10^{-4} µCi/cc
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	≤ 4.32×10^{-5} µCi/cc	≤ 5.19×10^{-5} µCi/cc
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 55 gpm	≤ 62.1 gpm
b. Δ Flow Timer	≤ 45 seconds	≤ 47 seconds
c. Equipment Area Temperature - High		
1. Heat Exchanger Room	≤ 98.5°F	≤ 101.5°F
2. Pump Rooms A & B	≤ 165°F	≤ 169.5°F
3. Valve Nest Room	≤ 110°F	≤ 114.5°F
4. Demineralizer Rooms 1 and 2	≤ 110°F	≤ 114.5°F
5. Receiving Tank Room	≤ 110°F	≤ 114.5°F

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Cont'd)		
d. Equipment Area Δ Temperature - High		
1. Heat Exchanger Room	≤ 33°F	≤ 36.5°F
2. Pump Rooms A and B	≤ 78°F	≤ 82°F
3. Valve Nest Room	≤ 46°F	≤ 49.5°F
4. Demineralizer Rooms 1 and 2	≤ 46°F	≤ 49.5°F
5. Receiving Tank Room	≤ 46°F	≤ 49.5°F
e. Reactor Vessel Water Level - Low Low Level 2	≥ - 43 inches*	≥ - 47 inches
f. Main Steam Line Tunnel Ambient Temperature - High	≤ 135°F	≤ 142.5°F
g. Main Steam Line Tunnel Δ Temperature - High	≤ 51°F	≤ 55°F
h. SLCS Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 222" H_2O**	≤ 230.5" H_2O**
b. RCIC Steam Line Flow - High Timer	≥ 3 seconds	≤ 13 seconds
c. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 55 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
e. RCIC Equipment Room Ambient Temperature - High	≤ 182°F	≤ 186.4°F
f. RCIC Equipment Room Δ Temperature - High	≤ 96°F	≤ 99°F

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Cont'd)		
g. Main Steam Line Tunnel Ambient Temperature - High	≤ 135°F	≤ 142.5°F
h. Main Steam Line Tunnel Δ Temperature - High	≤ 51°F	≤ 55°F
i. Main Steam Line Tunnel Temperature Timer	0 seconds	NA
j. RHR Equipment Room Ambient Temperature - High	≤ 117°F	≤ 121.1°F
k. RHR Equipment Room Δ Temperature - High	≤ 29°F	≤ 33.6°F
l. RHR/RCIC Steam Line Flow - High	≤ 156" H_2O**	≤ 164.5" H_2O**
m. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
n. Manual Initiation	NA	NA
6. RHR SYSTEM ISOLATION		
a. RHR Equipment Area Ambient Temperature - High	≤ 117°F	≤ 121.1°F
b. RHR Equipment Area Δ Temperature - High	≤ 29°F	≤ 33.6°F
c. Reactor Vessel Water Level - Low Level 3	≥ 9.7 inches*	≥ 8.7 inches
d. Reactor Vessel Water Level - Low Low Low Level 1	≥ - 143 inches*	≥ - 147 inches

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. RHR SYSTEM ISOLATION (Cont'd)		
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 135 psig	≤ 150 psig
f. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
7. MANUAL INITIATION	NA	NA

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

**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

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TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$
b. Drywell Pressure - High	$\leq 10^{(a)}$
c. Containment Purge Isolation Radiation - High$^{(b)}$	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low Level 1	≤ 1.0 */$\leq 10^{(a)}$**
b. Main Steam Line Radiation - High$^{(b)}$	≤ 1.0 */$\leq 10^{(a)}$**
c. Main Steam Line Pressure - Low	≤ 1.0 */$\leq 10^{(a)}$**
d. Main Steam Line Flow - High	≤ 0.5 */$\leq 10^{(a)}$**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Area Temperature - High (Turbine Bldg)	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$
b. Drywell Pressure - High	$\leq 10^{(a)}$
c. Fuel Building Ventilation Exhaust Radiation - High$^{(b)}$	NA
d. Reactor Building Annulus Ventilation Exhaust Radiation - High$^{(b)}$	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq 10^{(a)}$##
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	$\leq 10^{(a)}$###
b. RCIC Steam Line Flow-High Timer	NA
c. RCIC Steam Supply Pressure - Low	$\leq 10^{(a)}$
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
e. RCIC Equipment Room Ambient Temperature - High	NA
f. RCIC Equipment Room Δ Temperature - High	NA
g. Main Steam Line Tunnel Ambient Temperature - High	NA
h. Main Steam Line Tunnel Δ Temperature - High	NA

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TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
i. Main Steam Line Tunnel Temperature Timer	NA
j. RHR Equipment Room Ambient Temperature - High	NA
k. RHR Equipment Room Δ Temperature - High	NA
l. RHR/RCIC Steam Line Flow - High	NA
m. Drywell Pressure - High	NA
n. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. Reactor Vessel Water Level - Low Level 3	$\leq 10^{(a)}$
d. Reactor Vessel Water Level - Low Low Low Level 1	$\leq 10^{(a)}$
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
7. MANUAL INITIATION	NA

(a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays.

(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 for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.5.3-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

##Time delay of 45-47 seconds.

###Time delay of 0-13 seconds.

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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 - Low Low Level 2	S	M	$R^{(b)}$	1, 2, 3
b. Drywell Pressure - High	S	M	$R^{(b)}$	1, 2, 3
c. Containment Purge Isolation Radiation - High	S	M	R	1, 2, 3
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	$R^{(b)}$	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	$R^{(b)}$	1
d. Main Steam Line Flow - High	S	M	$R^{(b)}$	1, 2, 3
e. Condenser Vacuum - Low	S	M	$R^{(b)}$	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Area Temperature-High (Turbine Building)	S	M	$R^{(b)}$	1, 2, 3

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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. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low Level 2	S	M	R(b)	1, 2, 3
b. Drywell Pressure - High	S	M	R(b)	1, 2, 3
c. Fuel Building Ventilation Exhaust Radiation - High	S	M	R	*
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	S	M	R	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Δ Flow Timer	NA	M	Q	1, 2, 3
c. Equipment Area Temperature - High	S	M	R	1, 2, 3
d. Equipment Area Δ Temperature - High	S	M	R	1, 2, 3
e. Reactor Vessel Water Level - Low Low Level 2	S	M	R(b)	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. SLCS Initiation	NA	M(a)	NA	1, 2, 3

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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. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	M	R(b)	1, 2, 3
b. RCIC Steam Line Flow-High Timer	NA	M	Q	1, 2, 3
c. RCIC Steam Supply Pressure - Low	S	M	R(b)	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R(b)	1, 2, 3
e. RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
f. RCIC Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
i. Main Steam Line Tunnel Temperature Timer	NA	M	Q	1, 2, 3
j. RHR Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
k. RHR Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
l. RHR/RCIC Steam Line Flow-High	S	M	R(b)	1, 2, 3
m. Drywell Pressure-High	S	M	R(b)	1, 2, 3
n. Manual Initiation	NA	R	NA	1, 2, 3

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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
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	S	M	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Level 3	S	M	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(b)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	M	R(b)	1, 2, 3
f. Drywell Pressure - High	S	M	R(b)	1, 2, 3
7. MANUAL INITIATION	NA	M	NA	1, 2, 3

*When handling irradiated fuel in the Fuel Building.

**When the reactor mode switch is in Run and/or any turbine stop valve is open.

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

(b) Calibrate trip unit setpoint at least once per 31 days.

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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. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status:

   1. Within 7 days, provided that the HPCS and RCIC systems are OPERABLE, or

   2. Within 72 hours, provided either the HPCS or the RCIC system is inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the following 24 hours.

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 At least once per 18 months, the ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit. Each test shall include at least one channel per trip system 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 trip system.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS 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. LPCS Pump Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	33
d. Reactor Vessel Pressure-Low (LPCS/LPCI Injection Valve Permissive)	4	1, 2, 3 4*, 5*	30 32
e. LPCI Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
f. LPCI Pump A Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	33
g. LPCS Pump Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/system	1, 2, 3, 4*, 5*	33
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low Level 1	2[b]	1, 2, 3	30
b. Drywell Pressure - High	2[b]	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low Level 3 (Permissive)	1	1, 2, 3	31
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	31
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	31
g. ADS Drywell Pressure Bypass Timer	2	1, 2, 3	31
h. ADS Manual Inhibit Switch	1	1, 2, 3	33
i. Manual Initiation	2/system	1, 2, 3	33

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
B. DIVISION II TRIP SYSTEM			
1. RHR B & C (LPCI MODE)			
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. Reactor Vessel Pressure-Low (LPCI Injection Valve Permissive)	4	1, 2, 3 4*, 5*	30 32
d. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
e. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	33
f. LPCI Pump C Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
g. Manual Initiation	1	1, 2, 3, 4*, 5*	33
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low Level 1	2[(b)]	1, 2, 3	30
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low Level 3 (Permissive)	1	1, 2, 3	31
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
f. ADS Drywell Pressure Bypass Timer	2	1, 2, 3	31
g. ADS Manual Inhibit Switch	1	1, 2, 3	33
h. Manual Initiation	2/system	1, 2, 3	33

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION III TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Level 2	4[b]	1, 2, 3, 4*, 5*	34[e]
b. Drywell Pressure - High	4[b]	1, 2, 3	34[e]
c. Reactor Vessel Water Level-High Level 8	2[c]	1, 2, 3, 4*, 5*	31
d. Condensate Storage Tank Level-Low	2[d]	1, 2, 3, 4*, 5*	35
e. Suppression Pool Water Level-High	2[d]	1, 2, 3, 4*, 5*	35
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	33
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	33
h. Manual Initiation	1	1, 2, 3, 4*, 5*	33

TRIP FUNCTION	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. LOSS OF POWER					
1. Divisions I and II					
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	3/bus	2/bus	3/bus	1, 2, 3, 4**, 5**	36
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	3/bus	2/bus	3/bus	1, 2, 3, 4**, 5**	36
2. Division III					
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	2/bus	1/bus	2/bus	1, 2, 3, 4**, 5**	37
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	36

See footnotes on next page

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TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip function in the tripped condition provided at least one other OPERABLE channel in the same trip function is monitoring that parameter.
(b) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS injection valves only.
(d) Provides signal to open HPCS suppression pool suction valve only.
(e) This trip function logic is one-out-of-two taken twice. Therefore, each one-out-of-two logic is defined as a separate trip system for HPCS when complying with ACTION 34.
* When the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when ESF equipment is required to be OPERABLE.
# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 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 Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system inoperable.
b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS valve or ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. For one trip system, place that trip system in the tripped condition within one hour* or declare the HPCS system inoperable.
b. For both trip systems, declare the HPCS system inoperable.

ACTION 35 - With the number of OPERABLE channels 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* or declare the HPCS system inoperable.

Action 36 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour*; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST. Otherwise, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 37 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

---

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION I TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low Level 1	$\geq$-143 inches*	$\geq$-147 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. LPCS Pump Discharge Flow-Low	$\geq$ 875 gpm	$\geq$ 750 gpm
d. Reactor Vessel Pressure-Low (LPCS/LPCI Injection Valve Permissive)	$\geq$ 487 psig	$\geq$ 472 psig, $\leq$ 502 psig
e. LPCI Pump A Start Time Delay Relay	$\leq$ 7 seconds	7 ± 0.7 seconds
f. LPCI Pump A Discharge Flow-Low	$\geq$ 1100 gpm	$\geq$ 900 gpm
g. LPCS Pump Start Time Delay Relay	$\leq$ 2 seconds	2 ± 0.2 seconds
h. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low Level 1	$\geq$-143 inches*	$\geq$-147 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low Level 3	$\geq$ 9.7 inches*	$\geq$ 8.7 inches
e. LPCS Pump Discharge Pressure-High	$\geq$ 145 psig, increasing	$\geq$ 130 psig, increasing
f. LPCI Pump A Discharge Pressure-High	$\geq$ 135 psig, increasing	$\geq$ 120 psig, increasing
g. ADS Drywell Pressure Bypass Timer	$\leq$ 5 minutes	$\leq$ 5.5 minutes
h. ADS Manual Inhibit Switch	NA	NA
i. Manual Initiation	NA	NA

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
B. DIVISION II TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low Level 1	$\geq$-143 inches*	$\geq$-147 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Reactor Vessel Pressure-Low (LPCI Injection Valve Permissive)	$\geq$ 487 psig	$\geq$ 472 psig, $\leq$ 502 psig
d. LPCI Pump B Start Time Delay Relay	$\leq$ 7 seconds	7 ± 0.7 seconds
e. LPCI Pump Discharge Flow-Low	$\geq$ 1100 gpm	$\geq$ 900 gpm
f. LPCI Pump C Start Time Delay Relay	$\leq$ 2 seconds	2 ± 0.2 seconds
g. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low Level 1	$\geq$-143 inches*	$\geq$-147 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low Level 3	$\geq$ 9.7 inches*	$\geq$ 8.7 inches
e. LPCI Pump B and C Discharge Pressure-High	$\geq$ 135 psig, increasing	$\geq$ 120 psig, increasing
f. ADS Drywell Pressure Bypass Timer	$\leq$ 5 minutes	$\leq$ 5.5 minutes
g. ADS Manual Inhibit Switch	NA	NA
h. Manual Initiation	NA	NA

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION III TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - (Low Low Level 2)	$\geq$-43 inches*	$\geq$-47 inches
b. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
c. Reactor Vessel Water Level - High Level 8	$\leq$ 51 inches*	$\leq$ 55 inches
d. Condensate Storage Tank Level - Low	$\geq$ 0 inches**	$\geq$ -4.5 inches**
e. Suppression Pool Water Level - High	$\leq$ 7 inches#	$\leq$ 8 inches#
f. Pump Discharge Pressure - High	$\geq$ 145 psig	$\geq$ 120 psig
g. HPCS System Flow Rate - Low	$\geq$ 625 gpm	$\geq$ 500 gpm
h. Manual Initiation	NA	NA
D. LOSS OF POWER		
1. Divisions I and II		
a. 4.16 kv Emergency Bus Undervoltage (Sustained Undervoltage)##	a. 4.16 kv Basis - 2970 ± 60 volts	2970 ± 120 volts
	b. 3 ± 0.3 sec. time delay	3 ± 0.33 sec. time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3740 ± 75 volts	3740 ± 135 volts
	b. 60 ± 6 sec. time delay (w/o LOCA)	60 ± 6.6 sec. time delay
	3 ± 0.3 sec. time delay (w/LOCA)	3 ± 0.33 sec. time delay

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TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (continued)		
2. Division III		
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	a. 4.16 kv Basis - 3045 ± 153 volts	3045 ± 214 volts
	b. 3 ± 0.3 sec. time delay	3 ± 0.33 sec. time delay
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - 3777 ± 30 volts	3777 ± 75 volts
	b. 60 ± 6 sec. time delay (w/o LOCA)	60 ± 6.6 sec. time delay
	c. 3 ± 0.3 sec. time delay (w/LOCA)	3 ± 0.33 sec. time delay

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

**(Bottom of CST is at EL 95'1".) The levels are measured from the instrument zero level of EL 96'6".

#(Bottom of suppression pool is at EL 70'.) The levels are measured from the instrument zero level of EL 89'9".

##These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

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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	≤ 37
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
a. Pumps A and B	≤ 37
b. Pump C	≤ 37
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 27
5. LOSS OF POWER	NA

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R[(a)]	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R[(a)]	1, 2, 3
c. LPCS Pump Discharge Flow-Low	S	M	R[(a)]	1, 2, 3, 4*, 5*
d. Reactor Vessel Pressure-Low (LPCS/LPCI Injection Valve Permissive)	S	M	R[(a)]	1, 2, 3, 4*, 5*
e. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
f. LPCI Pump A Discharge Flow-Low	S	M	R[(a)]	1, 2, 3, 4*, 5*
g. LPCS Pump Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R[(a)]	1, 2, 3
b. Drywell Pressure-High	S	M	R[(a)]	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low Level 3	S	M	R[(a)]	1, 2, 3
e. LPCS Pump Discharge Pressure-High	S	M	R[(a)]	1, 2, 3
f. LPCI Pump A Discharge Pressure High	S	M	R[(a)]	1, 2, 3
g. ADS Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
h. ADS Manual Inhibit Switch	NA	M	NA	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3

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 FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION II TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Pressure-Low (LPCI Injection Valve Permissive)	S	M	R(a)	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	S	M	R(a)	1, 2, 3, 4*, 5*
f. LPCI Pump C Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
g. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(a)	1, 2, 3
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low Level 3	S	M	R(a)	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	S	M	R(a)	1, 2, 3
f. ADS Drywell Pressure Bypass Timer	NA	M	Q	1, 2, 3
g. ADS Manual Inhibit Switch	NA	M	NA	1, 2, 3
h. Manual Initiation	NA	R	NA	1, 2, 3

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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 FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION III TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Level 2	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	S	M	R(a)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R(a)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(a)	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	S	M	R(a)	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	S	M	R(a)	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Divisions I and II				
a. 4.16 kv Standby Bus Under-voltage (Sustained Under-voltage)	S	M	R	1, 2, 3, 4**, 5**
b. 4.16 kv Standby Bus Under-voltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**
2. Division III				
a. 4.16 kv Standby Bus Under-voltage (Sustained Under-voltage)	S	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kv Standby Bus Under-voltage (Degraded Voltage)	S	M	R	1, 2, 3, 4**, 5**

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required when ESF equipment is required to be OPERABLE.
(a) Calibrate trip unit setpoint at least once per 31 days.

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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 channels 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-RPT system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip 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, restore the inoperable channel to OPERABLE status within 30 days or be in at least STARTUP within the next 6 hours.

c. Otherwise, restore at least one inoperable channel in each trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS-RPT system 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 4.3.4.1-1.

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 CHANNELS PER TRIP SYSTEM [a]
1. Reactor Vessel Water Level - Low Low Level 2	2
2. Reactor Vessel Pressure - High	2

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

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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	$\geq$-43 inches*	$\geq$-47 inches
2. Reactor Vessel Pressure - High	$\leq$ 1127 psig	$\leq$ 1135 psig

*See Bases Figure B3/4 3-1.

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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	S	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 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 40% 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, restore the inoperable trip system to OPERABLE status within 72 hours or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 6 hours.

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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 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME 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) shown in Table 3.3.4.2-3 such that both types of channel inputs are tested at least once per 36 months. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limits.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured at least once per 60 months.

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) A 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) This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 187* psig, equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

*To allow for instrumentation accuracy, calibration and drift, a setpoint of $\leq$ 177 psig turbine first stage pressure shall be used.

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TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure	$\leq$5% closed	$\leq$ 7% closed
2. Turbine Control Valve-Fast Closure	$\geq$ 530 psig	$\geq$ 465 psig

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TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Milliseconds)
1. Turbine Stop Valve-Closure	$\leq$ 140
2. Turbine Control Valve-Fast Closure	$\leq$ 140

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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*(a)	R#(a)
2. Turbine Control Valve-Fast Closure	M*(a)	R#(a)

---

*Including trip system logic testing.

#Calibrate the first stage pressure transmitter trip unit setpoint at least once per 31 days.

(a) The CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

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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 150 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 FUNCTION[(a)]	ACTION
1. Reactor Vessel Water Level - Low Low Level 2	4[(b)]	50
2. Reactor Vessel Water Level - High Level 8	2[(c)]	51
3. Condensate Storage Tank Water Level - Low	2[(d)]	52
4. Suppression Pool Water Level - High	2[(d)]	52
5. Manual Initiation	1[(e)]	53

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

(b) This trip function logic is one-out-of-two taken twice. Therefore, each one-out-of-two logic is defined as a separate trip system for RCIC when complying with ACTION 50.

(c) One trip system with two-out-of-two logic.

(d) One trip system with one-out-of-two logic.

(e) One trip system with one channel.

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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 Function requirement:

a. For one trip system, place that trip system 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 Function requirement, declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels 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 or declare the RCIC system inoperable.

ACTION 53 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.

*The provisions of Specification 3.0.4 are not applicable.

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TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low Level 2	≥ -43 inches*	≥ -47 inches
2. Reactor Vessel Water Level - High Level 8	≤ 51 inches*	≤ 55 inches
3. Condensate Storage Tank Level - Low	≥ 0 inches	≥ -4.5 inches
4. Suppression Pool Water Level - High	≤ 6.5 inches	≤ 8 inches
5. 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(a)
b. Reactor Vessel Water Level - High Level 8	S	M	R(a)
c. Condensate Storage Tank Level - Low	S	M	R(a)
d. Suppression Pool Water Level - High	S	M	R(a)
e. Manual Initiation	NA	R	NA

(a) Calibrate trip unit setpoint at least once per 31 days.

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3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod 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 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.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod 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 BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD PATTERN CONTROL SYSTEM			
a. Low Power Setpoint	2	1, 2	60
b. High Power Setpoint	2	1, 2	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	6	1	61
b. Inoperative	6	1, 2, 5	61
c. Downscale	6	1	61
d. Neutron Flux - Upscale, Startup	6	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in[(a)]	3	2	61
	2**	5	62
b. Upscale[(b)]	3	2	61
	2**	5	62
c. Inoperative[(b)]	3	2	61
	2**	5	62
d. Downscale[(c)]	3	2	61
	2**	5	62
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in[(d)]	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale[(d)]	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5*	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62

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ACTION

ACTION 60 - Declare the RPCS inoperable and take the ACTION required by Specification 3.1.4.2.

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

** OPERABLE channels must be associated with SRM required OPERABLE per Specification 3.9.2.

(a) This function shall be automatically bypassed if detector count rate is $\geq$ 100 cps or the IRM channels are on range 3 or higher.

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

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

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	27.5 ± 3% of RATED THERMAL POWER	27.5 ± 7.5% of RATED THERMAL POWER
b. High Power Setpoint	62.5 ± 3% of RATED THERMAL POWER	62.5 ± 7.5% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - 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	$\leq 1 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 0.7 cps	$\geq$ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 division of full scale	$\leq$ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 division of full scale	$\geq$ 3/125 division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High - LISN602A	$\leq$ 18.00"	$\leq$ 21.12"
LISN602B	$\leq$ 18,00"	$\leq$ 21.60"
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108% of rated flow	$\leq$ 111% of rated flow

*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.

**Provided signal to noise ratio is $\geq$ 2, otherwise setpoint of 3 cps and allowable 1.8 cps.

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TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	S[(f)]	S/U[(b)(e)] D[(c)], M[(d)(e)]	SA#	1, 2
b. High Power Setpoint	S[(f)]	S/U[(b)(e)] D[(c)], M[(d)(e)]	SA#	1, 2
2. APRM				
a. Flow Biased Neutron Flux - Upscale	NA	S/U[(b)], M	SA	1
b. Inoperative	NA	S/U[(b)], M	NA	1, 2, 5
c. Downscale	NA	S/U[(b)], M	SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U[(b)], M	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)], W	NA	2, 5
b. Upscale	NA	S/U[(b)], W	SA	2, 5
c. Inoperative	NA	S/U[(b)], W	NA	2, 5
d. Downscale	NA	S/U[(b)], W	SA	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[(b)], W	NA	2, 5
b. Upscale	NA	S/U[(b)], W	SA	2, 5
c. Inoperative	NA	S/U[(b)], W	NA	2, 5
d. Downscale	NA	S/U[(b)], W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	M	R#	1, 2, 5*
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[(b)], M	SA	1

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TABLE 4.3.6-1 (Continued)

CONTROL ROD 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. Within one hour prior to control rod movement, unless performed within the previous 24 hours, and as each power range above the RPCS low power setpoint is entered for the first time during any 24 hour period during power increase or decrease.

d. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

e. Includes reactor manual control multiplexing system input.

f. Verify the Turbine Bypass valves are closed when THERMAL POWER is greater than 20% 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.

\# Calibrate trip unit setpoint once per 31 days.

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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.

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TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Ventilation Radiation Monitor				
a. Local Intake	2	1,2,3,5 and *	$\leq 0.97 \times 10^{-5}$ µCi/cc(a)	70
b. Remote Intake	1	1,2,3,5 and *	$\leq 0.97 \times 10^{-5}$ µCi/cc(a)	71
2. Area Monitor				
a. Fuel Building Spent Fuel Storage Pool	1	#	≤ 15 mR/hr(a)	71
3. Main Condenser Offgas Post-Treatment System Effluent Monitoring System				
a. Noble Gas Activity Monitor - (Providing Alarm and Automatic Termination of Release)	1	**	$\leq 5.08 \times 10^{5}$ cpm	72
4. Condenser Air Ejector Pretreatment Radioactivity Monitor (Prior to Input to Holdup System)				
a. Noble Gas Activity Monitor	1	**	$\leq 2.48 \times 10^{4}$ mR/hr(a)	73

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TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATION

*When irradiated fuel is being handled in the primary containment or the Fuel Building.

**During operation of the main condenser air ejector.

(a)Alarm only.

#With fuel in the spent fuel storage pool.

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ACTION

ACTION 70 -

a. With one of the required monitors inoperable, restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room air conditioning system in the emergency mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room air conditioning system in the emergency mode of operation within one hour.

ACTION 71 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 72 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 73 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases may continue to the environment for up to 72 hours provided:

a. The offgas system is not bypassed, and

b. At least one post treatment noble gas activity effluent monitor is OPERABLE;

Otherwise, be in at least HOT STANDBY within 12 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. Main Control Room Ventilation Radiation Monitor				
a. Local Intake	S	M	R	1, 2, 3, 5 and *
b. Remote Intake	S	M	R	1, 2, 3, 5 and *
2. Area Monitor				
a. Fuel Building Spent Fuel Storage Pool	S	M	R	#
3. Main Condenser Offgas Post-Treatment System Effluent Monitoring System				
a. Noble Gas Activity Monitor - (Providing Alarm and Automatic Termination of Release)	D	Q	R	**
4. Condenser Air Ejector Pretreatment Radioactivity Monitor				
a. Noble Gas Activity Monitor	D	Q	R	**

*When irradiated fuel is being handled in the primary containment or the Fuel Building.
**During operation of the main condenser air ejector.
#With fuel in the spent fuel storage pool.

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SEISMIC MONITORING INSTRUMENTATION

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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 of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit within the next 10 days a Special Report to the Commission, pursuant to Specification 6.9.2, 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 greater than or equal to 0.01g 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. Within 10 days, a Special Report shall be prepared and submitted to the Commission, pursuant to Specification 6.9.2, describing the magnitude, frequency spectrum and resultant effect upon unit features important to safety.

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TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. Reactor Bldg Mat EL 70'0"	0 ± 1.0 g	1
b. Reactor Bldg Ext Shield Wall EL 232'0"	0 ± 1.0 g	1
c. Reactor Bldg Drywell EL 151'0"	0 ± 1.0 g	1
d. Free Field - Grade Level	0 ± 1.0 g	1
2. Triaxial Peak Accelerographs		
a. Reactor Bldg SLCS Storage Tank	0 ± 10.0 g	1
b. Reactor Bldg - RHR Inj. Piping	0 ± 10.0 g	1
c. Aux. Bldg Service Water Piping	0 ± 10.0 g	1
3. Triaxial Seismic Switches		
a. Reactor Bldg Mat EL 70'0"	0.025 to 0.25 g	1(a)
4. Triaxial Response-Spectrum Recorders		
a. Reactor Bldg Mat EL 70'0"	0 ± 2 g	1(a)
b. Reactor Bldg Floor EL 141'0"	0 ± 2 g	1
c. Auxiliary Bldg Mat EL 70'0"	0 ± 2 g	1
d. Auxiliary Bldg Floor EL 141'0"	0 ± 2 g	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. Reactor Bldg. Mat EL 70'0"	M	SA	R
b. Reactor Bldg. Ext Shield Wall EL 232'0"	M	SA	R
c. Reactor Bldg. Drywell EL 151'0"	M	SA	R
d. Free Field-Grade Level	M	SA	R
2. Triaxial Peak Accelerographs			
a. Reactor Bldg. SLCS Storage Tank	NA	NA	R
b. Reactor Bldg. - RHR Inj. Piping	NA	NA	R
c. Aux. Bldg. Service Water Piping	NA	NA	R
3. Triaxial Seismic Switches			
a. Reactor Bldg. Mat EL 70'0"	M[(a)]	SA	R
4. Triaxial Response-Spectrum Recorders			
a. Reactor Bldg. Mat EL 70'0'	M	SA	R
b. Reactor Bldg. Floor EL 141'0"	NA	NA	R
c. Auxiliary Bldg. Mat EL 70'0"	NA	NA	R
d. Auxiliary Bldg. Floor EL 141'0"	NA	NA	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, prepare and submit within the next 10 days a Special Report to the Commission, pursuant to Specification 6.9.2, 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.

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TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
a. Wind Speed	
1. Elev. 30 ft.	1
2. Elev. 150 ft.	1
b. Wind Direction	
1. Elev. 30 ft.	1
2. Elev. 150 ft.	1
c. Air Temperature Difference	
1. Elev. 30/150 ft.	1

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TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 30 ft.	D	SA
2. Elev. 150 ft.	D	SA
b. Wind Direction		
1. Elev. 30 ft.	D	SA
2. Elev. 150 ft.	D	SA
c. Air Temperature Difference		
1. Elev. 30/150 ft.	D	SA

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REMOTE SHUTDOWN MONITORING INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown monitoring instrumentation channels and controls shown in Table 3.3.7.4-1 and 3.3.7.4-2 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required by Table 3.3.7.4-2, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 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.

4.3.7.4.2 Each of the above required remote shutdown system control circuits shall be demonstrated OPERABLE by verifying, at least once per 18 months, its capability to perform its intended function(s).

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TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION*	MINIMUM CHANNELS OPERABLE PER PANEL
1. Reactor Vessel Pressure	RSP1, RSP2	1
2. Reactor Vessel Water Level	RSP1, RSP2	1
3. Safety/Relief Valve Demand Position, 3 valves	RSP1, RSP2	1/valve
4. Suppression Pool Water Level	RSP1, RSP2	1
5. Suppression Pool Water Temperature	RSP1, RSP2	1
6. Drywell Pressure	RSP1, RSP2	1
7. Drywell Temperature	RSP1, RSP2	1
8. RHR System Flow: Loop A	RSP1	1
Loop B	RSP2	1
Loop C	RSP2	1
9. RHR Hx Cooling Water System Flow: Loop A	RSP1	1
Loop B	RSP2	1
10. RCIC System Flow	RSP1	1
11. RCIC Turbine Speed	RSP1	1

*RSP1 - Remote Shutdown Panels 1C61*P001 and 1RSS*PNL101
RSP2 - Remote Shutdown Panel 1RSS*PNL102

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TABLE 3.3.7.4-2

REMOTE SHUTDOWN SYSTEM CONTROLS

		MINIMUM CHANNELS OPERABLE RSP1	RSP2
1.	RCIC Suction from CST MOV (1E51*MOVF010)	1	NA
2.	RCIC Injection Shutoff MOV (1E51*MOVF013)	1	NA
3.	RCIC Min. Flow to Suppression Pool MOV (1E51*MOVF019)	1	NA
4.	RCIC Test Bypass to CST MOV (1E51*MOVF022)	1	NA
5.	RCIC Gland Seal Air Compressor (1E51-C002C)	1	NA
6.	RCIC Pump Suction from Suppression Pool MOV (1E51*MOVF031)	1	NA
7.	RCIC Steam to Turbine MOV (1E51*MOVF045)	1	NA
8.	RCIC Turbine Lube Oil Cooling MOV (1E51*MOVF046)	1	NA
9.	RCIC Test Bypass to CST MOV (1E51*MOVF059)	1	NA
10.	RCIC Steam Supply Inboard Isolation MOV(1E51*MOVF063)	1	NA
11.	RCIC Steam Supply Outboard Isolation MOV(1E51*MOVF064)	1	NA
12.	RCIC Turbine Exhaust to Suppression Pool MOV(1E51*MOVF068)	1	NA
13.	RCIC Steam Line Warmup Line Isolation MOV(1E51*MOVF076)	1	NA
14.	RCIC Vacuum Breaker Outboard Isolation MOV(1E51*MOVF077)	1	NA
15.	RCIC Vaccum Breaker Inboard Isolation MOV(1E51*MOVF078)	1	NA
16.	RCIC Turbine Flow Controller (1C61*FICR001)	1	NA
17.	RCIC Turbine Trip & Throttling MOV (1E51*MOVC002)	1	NA
18.	RCIC Turbine Local Control Select Switch (1C61A-S11)	1	NA
19.	RHR Pump (1E12*PC002A, 2B, 2C)	1	2[(a)]
20.	RHR Hx Shell Side Outlet MOV (1E12*MOVF003A, B)	1	1
21.	RHR Pump Suction MOV (1E12*MOVF004A, B; 1E12*MOVF105)	1	2[(a)]

(a) One per control equipment

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TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

		MINIMUM CHANNELS OPERABLE RSP1	MINIMUM CHANNELS OPERABLE RSP2
22.	RHR Shutdown Cooling MOV (1E12*MOVF006A, 6B)	$2^{(a)}$	NA
23.	RHR Outboard Shutdown Isolation MOV (1E12*MOVF008)	1	NA
24.	RHR Inboard Shutdown Isolation MOV (1E12*MOVF009)	1	NA
25.	RHR Hx Flow to Suppression Pool MOV (1E12*MOVF011A, B)	1	1
26.	RHR Reactor Head Spray MOV (1E12*MOVF023)	1	NA
27.	RHR Test Line MOV (1E12*MOVF024A, B)	1	1
28.	RHR Hx Flow to RCIC MOV (1E12*MOVF026A)	1	NA
29.	RHR Injection Shutoff MOV (1E12*MOVF027A, B)	1	1
30.	RHR Upper Pool Cooling Shutoff MOV (1E12*MOVF037A, B)	1	1
31.	RHR Injection MOV (1E12*MOVF042A, B, C)	1	$2^{(a)}$
32.	RHR Hx Shell Side Inlet MOV (1E12*MOVF047A, B)	1	1
33.	RHR Hx Shell Side Bypass MOV (1E12*MOVF048A, B)	1	1
34.	RHR Discharge to Radwaste MOV (1E12*MOVF040)	1	NA
35.	RHR Steam Isolation MOV (1E12*MOVF052A, B)	1	1
36.	RHR Injection MOV (1E12*MOVF053A, B)	1	1
37.	RHR Pump Minimum Flow MOV (1E12*MOVF064A, B, C)	1	$2^{(a)}$
38.	RHR Hx Water Discharge MOV (1E12*MOVF068A, B)	1	1
39.	Safety Relief Valves (1B21*RVF051C, G, D)	$3^{(a)}$	$3^{(a)}$
40.	SSW Pump (1SWP*P2A, 2C,[(b)] 2B, 2D)	1	$2^{(a)}$
41.	Normal Service Water Isolation MOV (1SWP*MOV96A, B)	1	1
42.	SSW Cooling Tower Inlet MOV (1SWP*MOV55A, B)	1	1
43.	SSW Component Cooling Water Inlet MOV (1SWP*MOV510A, B)	1	1
44.	SSW Component Cooling Water Outlet MOV (1SWP*MOV504A, B)	1	1

(a) One per control equipment.
(b) SSW pump 1SWP*P2C is provided on panel 1EGS*PNL4C.

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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 Demand Position	M	NA
4. Suppression Pool Water Level	M	R
5. Suppression Pool Water Temperature	M	R
6. Drywell Pressure	M	R
7. Drywell Temperature	M	R
8. RHR System Flow: Loop A	M	R
Loop B	M	R
Loop C	M	R
9. RHR Hx Cooling Water System Flow: Loop A	M	R
Loop B	M	R
10. RCIC System Flow	M	R
11. RCIC Turbine Speed	M	R

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required 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

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERABLE CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2	80
2. Reactor Vessel Water Level				
a. Wide Range	2	1	1,2	80
b. Fuel Zone	2	1	1,2	80
3. Suppression Pool Water Level	2	1	1,2	80
4. Suppression Pool Water Temperature	2/sector	1/sector	1,2	80
5. Primary Containment Pressure	2	1	1,2	80
6. Drywell Pressure	2	1	1,2	80
7. Drywell Air Temperature	2	1	1,2	80
8. Drywell and Primary Containment Hydrogen Concentration Analyzer and Monitor	2	1	1,2	80
9. Area Radiation#				
a. Primary Containment Area	2	1	1,2,3	81
b. Drywell Area	2	1	1,2,3	81

#High range gross gamma monitors.

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Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATIONS

ACTION STATEMENTS

Action 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 81 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

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

b. Prepare and submit, within 14 days following the event, a Special Report to the Commission, pursuant to Specification 6.9.2, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1, 2
2. Reactor Vessel Water Level			
a. Wide Range	M	R	1, 2
b. Fuel Zone	M	R	1, 2
3. Suppression Pool Water Level	M	R	1, 2
4. Suppression Pool Water Temperature	M	R	1, 2
5. Primary Containment Pressure	M	R	1, 2
6. Drywell Pressure	M	R	1, 2
7. Drywell Air Temperature	M	R	1, 2
8. Drywell and Primary Containment Hydrogen Concentration Analyzer and Monitor	M	Q*	1, 2
9. Area Radiation#			
a. Primary Containment Area	M	R**	1, 2, 3
b. Drywell Area	M	R**	1, 2, 3

*Using sample gas containing:
a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

**The CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

#High range gross gamma monitors.

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SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three,

b. In OPERATIONAL CONDITION 3 and 4, two.

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 OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable 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 0.7 cps# with the detector fully inserted.

*With IRM's on range 2 or below.

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

#Provided the signal to noise ratio $\geq$2, otherwise 3 cps.

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

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

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

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

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 Within 72 hours prior to use, when required for the LPRM calibration functions, the traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs.

*Only the detector(s) in the location(s) of interest are required to be OPERABLE.

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FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.8-1 shall be OPERABLE.

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

ACTION:

a. With any, but no more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3.7.8-1 inoperable, restore the inoperable Function A instrument(s) to OPERABLE status within 14 days or, within the next hour, establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour. If the instrument(s) is located inside the containment, inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.8 and 4.6.2.6.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3.7.8-1 inoperable, or with any Function B fire detection instruments shown in Table 3.3.7.8-1 inoperable, or with any two or more adjacent instruments shown in Table 3.3.7.8-1 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour. If the instrument(s) is located inside the containment, inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.8 and 4.6.2.6.

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

SURVEILLANCE REQUIREMENTS

4.3.7.8.1 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.

4.3.7.8.2 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.

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TABLE 3.3.7.8-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		TOTAL INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
I. CONTROL BUILDING ZONE				
SD-1	HVAC ROOM, EL 115'0" & 116'0"			6/0
SD-2	HPCS SWGR, EL 115'0" & 116'0"			3/0
SD-3	BATTERY ROOMS (3) & DC EQUIP RMS, EL 115'0" & 116'0"			8/0
SD-4	HVAC ROOM, EL 98'0"			6/0
SD-5	STBY SWGR ROOM B, EL 98'0"			3/0
SD-6	STBY SWGR ROOM A, EL 98'0"			3/0
SD-15	HVAC ROOM 1A, EL 70'0"			2/0
SD-16	HVAC ROOM 1B, EL 70'0"			2/0
SD-17	CABLE VAULT, EL 70'0"			0/3
SD-18	CABLE VAULT, EL 70'0"			0/4
SD-19	CABLE VAULT, EL 70'0"			0/9
SD-20	CABLE CHASES, EL 70'0"			17/0
SD-50	CABLE CHASES, EL 98'0"			9/0
SD-54	CABLE CHASES, EL 116'0"			10/0
SD-60	125 VDC SWGR & BATT CHGR, EL 115'0" & 116'0"			10/0
SD-61	GENERAL AREA, EL 98'0"			8/0
SD-125	PGCC PANEL MODULE, EL 136'0"	0/8		10/0
SD-126	PGCC PANEL MODULE, EL 136'0"	0/8		13/0
SD-127	PGCC PANEL MODULE, EL 136'0"	0/8		11/0
SD-128	PGCC PANEL MODULE, EL 136'0"	0/8		9/0
SD-129	PGCC PANEL MODULE, EL 136'0"	0/9		12/0
SD-130	PGCC PANEL MODULE, EL 136'0"	0/8		11/0
SD-131	PGCC PANEL MODULE, EL 136'0"	0/9		17/0
SD-132	PGCC PANEL MODULE, EL 136'0"	0/8		17/0
SD-133	PGCC PANEL MODULE, EL 136'0"	0/8		13/0
SD-134	PGCC PANEL MODULE, EL 136'0"	0/8		12/0
SD-135	PGCC PANEL MODULE, EL 136'0"	0/8		9/0
SD-136	PGCC PANEL MODULE, EL 136'0"	0/9		9/0
SD-137	PGCC PANEL MODULE, EL 136'0"	0/8		8/0
SD-138	PGCC PANEL MODULE, EL 136'0"	0/8		10/0
SD-139	PGCC PANEL MODULE, EL 136'0"	0/12		12/0
SD-140	PGCC PANEL MODULE, EL 136'0"	0/8		14/0
SD-141	PGCC PANEL MODULE, EL 136'0"	0/8		13/0
SD-142	PGCC PANEL MODULE, EL 136'0"	0/9		16/0

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

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TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		TOTAL INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
I. CONTROL BUILDING ZONE (Continued)				
SD-143	PGCC PANEL MODULE, EL 136'0"	0/9		17/0
SD-144	PGCC PANEL MODULE, EL 136'0"	0/9		17/0
SD-145	PGCC PANEL MODULE, EL 136'0"	0/8		8/0
SD-146	PGCC PANEL MODULE, EL 136'0"	0/8		8/0
SD-147	PGCC PANEL MODULE, EL 136'0"	0/12		14/0
SD-148	PGCC PANEL MODULE, EL 136'0"	0/12		18/0
SD-149	PGCC PANEL MODULE, EL 136'0"	0/10		14/0
SD-150	PGCC PANEL MODULE, EL 136'0"	0/9		15/0
SD-151	PGCC PANEL MODULE, EL 136'0"	0/10		10/0
SD-158	PGCC PANEL MODULE, EL 136'0"	0/8		8/0
SD-152	NON PANEL MODULE AREA NORTH, EL 135'0"			10/0
SD-153	NON PANEL AREA SOUTH, EL 135'0'"			10/0
SD-154	GENERAL AREA, EL 136'0'"			84/0
SD-162	REMOTE SHUTDOWN PANEL DIV I, EL 98'0"			1/0
SD-163	REMOTE SHUTDOWN PANEL DIV II, EL 98'0"			1/0
FD-26	CHARCOAL FILTER 1HVC*FLT3B, EL 115'0"	1/0		
FD-27	CHARCOAL FILTER 1HVC*FLT3A, EL 115'0"	1/0		
II. REACTOR BUILDING ZONE				
SD-57	#CONTAINMENT AREA, EL 114'0"			13/0
SD-102	ANNULUS AREA, EL 186'3"			28/0
SD-104	#CONTAINMENT AREA, EL 186'3"			17/0
SD-117	#CONTAINMENT AREA, EL 162'3"			7/0
SD-119	#CONTAINMENT AREA, EL 141'0"			13/0
SD-156	#CONTAINMENT AREA, EL 95'9"			2/0
FD-13	#RECIRC PUMPS - DRYWELL, EL 70'0" & 98'0"	2/0		

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

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

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TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		TOTAL INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
III. AUXILIARY BUILDING ZONE				
SD-28	HPCS PUMP ROOM EL 70'0"			1/0
SD-29	RHR PUMP ROOM B, EL 70'0"			2/0
SD-30	RHR PUMP ROOM C, EL 70'0"			2/0
SD-31	RHR PUMP ROOM A, EL 70'0"			2/0
SD-32	LPCS PUMP ROOM, EL 70'0"			1/0
SD-43	GENERAL AREA WEST, EL 95'0"			2/0
SD-49	GENERAL AREA, EL 141'0"			9/0
SD-52	GENERAL AREA EAST, EL 114'0"			5/0
SD-53	GENERAL AREA WEST, EL 114'0"			5/0
SD-55	PASS ROOM, EL 114'0"			1/0
SD-96	RCIC PUMP ROOM, EL 70'0"			0/2
SD-97	GENERAL AREA, EL 70'0"			4/0
SD-98	GENERAL AREA EAST, EL 95'9"			2/0
SD-99	GENERAL AREA WEST, EL 95'9"			2/0
SD-100	GENERAL AREA WEST, EL 95'9"			2/0
SD-101	STANDBY GAS TREATMENT ROOM "B", EL 141'0"			4/0
SD-103	STANDBY GAS TREATMENT ROOM "A", EL 141'0"			4/0
SD-106	ANNULUS MIXING FAN AREA, EL 171'0"			3/0
FD-33	STANDBY GAS TREATMENT FILTER "B", EL 141'0"	1/0		
FD-34	STANDBY GAS TREATMENT FILTER "A", EL 141'0"	1/0		
SD-164	WATER CURTAIN, EL 70'0"			0/2
SD-165	WATER CURTAIN, EL 141'0"			0/4
IV. FUEL BUILDING ZONE				
SD-33	FUEL POOL COOLING PUMP AREAS, EL 70'0"			2/0
SD-44	1ENS*SWG 3A & 4A AREA, EL 95'0"			7/0
SD-59	GENERAL AREA, EL 113'0"			13/0
SD-91	GENERAL AREA, EL 70'0"			7/0
SD-94	NEW FUEL RECEIVING AREA, EL 95'0"			2/0

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

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TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		TOTAL INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
IV. FUEL BUILDING ZONE (Continued)				
SD-110	FUEL POOL PURIFICATION & BACKWASH PUMP AREAS, EL 70'0"			3/0
SD-111	FUEL POOL COOLER (A & B) AREAS, EL 95'0"			2/0
SD-121	CHARCOAL FILTER "A" ROOM, EL 148'0"			2/0
SD-123	CHARCOAL FILTER "B" ROOM, EL 148'0"			2/0
SD-124	1RMS*CAB101 AREA, EL 148'0"			4/0
SD-155	GENERAL AREA, EL 113'0"			4/0
FD-35	CHARCOAL FILTER "A" ROOM, EL 148'0"	1/0		
FD-36	CHARCOAL FILTER "B" ROOM, EL 148'0"	1/0		
V. ELECTRICAL TUNNELS ZONE				
SD-79	GENERAL AREA, EL 67'6"			0/6
SD-80	GENERAL AREA, EL 67'6"			0/6
SD-81	GENERAL AREA, EL 67'6"			0/11
SD-82	GENERAL AREA, EL 67'6"			0/12
SD-83	GENERAL AREA, EL 70'0"			0/10
VI. PIPE TUNNEL ZONE				
SD-86	GENERAL AREA, EL 70'0"			0/9
SD-87	GENERAL AREA, EL 67'6"			0/4
SD-88	GENERAL AREA, EL 67'6"			0/5
SD-89	GENERAL AREA, EL 67'6"			0/8
VII. DIESEL GENERATOR BUILDING ZONE				
SD-105	GENERAL AREA, EL 98'0"			3/0
FD-16	DIESEL ROOM DIV. II, EL 98'0"	0/4		
FD-17	DIESEL ROOM DIV III, EL 98'0"	0/4		
FD-18	DIESEL ROOM DIV I, EL 98'0	0/4		

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

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TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		TOTAL INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
VIII. STANDBY SERVICE WATER PUMP HOUSE ZONE				
SD-72	STANDBY SERVICE WATER PUMP AREA			4/0
SD-73	STANDBY SERVICE WATER PUMP AREA			4/0

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

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LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.9 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. CHANNEL CHECK at least once per 24 hours,

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

c. CHANNEL CALIBRATION at least once per 18 months.

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.7.10-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.10-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in ACTION or explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

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

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.10-1.

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TABLE 3.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RMS-RE107)	1	100
2. Gross Radioactivity Monitors Providing Alarm but not Providing Automatic Termination of Release		
a. Cooling Tower Blowdown Line (RMS-RE108)	1	101
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (LWS-FE197)	1	102
b. Cooling Tower Blowdown Line (CWS-FE113)	1	102

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TABLE 3.3.7.10-1 (Continued)

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TABLE NOTATION

ACTION 100 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 101 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 102 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves generated in situ may be used to estimate flow.

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TABLE 4.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Line (RMS-RE107)	D	P	$R^{(3)}$	$Q^{(1)(2)}$
2. Gross Radioactivity Monitors Providing Alarm but not Providing Automatic Termination of Release				
a. Cooling Tower Blowdown Line (RMS-RE108)	D	M	$R^{(3)}$	$Q^{(2)}$
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line (LWS-FE197)	$D^{(4)}$	N.A.	R	Q
b. Cooling Tower Blowdown Line (CWS-FE113)	$D^{(4)}$	N.A.	R	Q

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TABLE 4.3.7.10-1 (Continued)

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TABLE NOTATION

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

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

b. Circuit failure.

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

a. Instrument indicates measured levels above the alarm setpoint.

b. Circuit failure.

c. Instrument indicates a downscale failure.

d. Instrument controls not set in operate mode.

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

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

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.7.11-1 shall be OPERABLE with their alarm/trip* setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3.7.11-1.

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.11-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

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

SUREVEILLANCE REQUIREMENTS

4.3.7.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.7.11-1.

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*The alarm/trip setpoints for the Explosive Gas Mixture in the Main Condenser Offgas Treatment System is set in accordance with Specification 3.11.2.6.

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TABLE 3.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Main Condenser Offgas Treatment System Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Analyzers (downstream of the recombiner)	1	**	123
2. Main Plant Exhaust Duct Monitoring System			
a. Noble Gas Activity Monitor	1	*	122
b. Iodine Sampler	1	*	124
c. Particulate Sampler	1	*	124
d. Effluent System Flow Rate Monitor	1	*	121
e. Sampler Flow Rate Monitor	1	*	121

TABLE 3.3.7.11-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. Fuel Building Exhaust Duct Monitoring System			
a. Noble Gas Activity Monitor	1	*	122
b. Iodine Sampler	1	*	124
c. Particulate Sampler	1	*	124
d. Flow Rate Monitor	1	*	121
e. Sampler Flow Rate Monitor	1	*	121
4. Radwaste Building Ventilation Exhaust Duct Monitoring System			
a. Noble Gas Activity Monitor	1	*	122
b. Iodine Sampler	1	*	124
c. Particulate Sampler	1	*	124
d. Flow Rate Monitor	1	*	121
e. Sampler Flow Rate Monitor	1	*	121

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TABLE 3.3.7.11-1 (Continued) FINAL DRAFT

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

*** During operation of the main condenser air ejector.

ACTION 121 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 122 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 123 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 124 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11.2.1.2-1.

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TABLE 4.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Main Condenser Offgas Treatment System Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion					
a. Hydrogen Analyzers (downstream of the recombiner)	D	NA	Q(3)	M	**
2. Main Plant Exhaust Duct Monitoring System					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Effluent System Flow Rate Monitor	D	NA	R	Q	*
e. Sampler Flow Rate Monitor	D	NA	R	Q	*

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TABLE 4.3.7.11-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. Fuel Building Exhaust Duct Monitoring System					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Flow Rate Monitor	D	NA	R	Q	*
e. Sampler Flow Rate Monitor	D	NA	R	Q	*
4. Radwaste Building Ventilation Exhaust Duct Monitoring System					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Flow Rate Monitor	D	NA	R	Q	*
e. Sampler Flow Rate Monitor	D	NA	R	Q	*

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TABLE 4.3.7.11-1 (Continued)

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TABLE NOTATIONS

* At all times.

** During main condenser offgas treatment system operation.

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

a. Instrument indicates measured levels above the alarm setpoint.

b. Circuit failure.

c. Instrument indicates a downscale failure.

d. Instrument controls not set in operate mode.

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

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

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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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:

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

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

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 cycling each of the following valves through at least one complete cycle from the running position:

   1) Four high pressure turbine stop valves,

   2) Four low pressure turbine intermediate stop valves,

   3) Four high pressure turbine control valves, and

   4) Four low pressure turbine intercept valves.

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

c. 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 excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

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LIMITING CONDITION FOR OPERATION

3.3.9 The plant systems actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and take the ACTION required by Table 3.3.9-1.

b. With one or more plant systems actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.9-1.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each plant system 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 4.3.9.1-1.

4.3.9.2 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 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B			
a. Drywell Pressure-High	2	1, 2, 3	150
b. Containment-To-Annulus ΔP High	3	1, 2, 3	151
c. Reactor Vessel Water Level-Low Low Low Level 1	2	1, 2, 3	150
d. Timers	1	1, 2, 3	152
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM			
a. Reactor Vessel Water Level-High Level 8	3	1	153

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TABLE 3.3.9-1 (Continued)

ACTION 150 - a. With one channel inoperable, place the inoperable channel in the tripped condition# within one hour or declare the associated system inoperable.

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 151 - a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

ACTION 152 - Declare the associated Containment Ventilation System inoperable.

ACTION 153 - a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

b. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

#Provided this does not actuate the system.

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TABLE 3.3.9-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B		
a. Drywell Pressure-High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
b. Containment-to-Annulus ΔP-High	-11.98 ± 0.22" H_2O	-11.98 + 0.31, - 0.27" H_2O
c. Reactor Vessel Water Level-Low Low Low Level 1	$\geq$-143 inches*	$\geq$-147 inches
d. Timer	600 ± 35 seconds	600 ± 45 seconds
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High Level 8	$\leq$ 51 inches*	$\leq$ 52.5 inches

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

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TABLE 4.3.9.1-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B				
a. Drywell Pressure-High	D	M	R[a]	1, 2, 3
b. Containment-to-Annulus ΔP-High	D	M	R[a]	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low Level 1	D	M	R[a]	1, 2, 3
d. Timer	NA	M	R	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High Level 8	D	M	R	1

[a] Calibrate trip unit setpoint once per 31 days.

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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 with:

a. Total core flow greater than or equal to 45% of rated core flow, or

b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION

a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.2):

   a) At least once per 8 hours, and

   b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

*See Special Test Exception 3.10.4.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

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4.4.1.1.1 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 at the hydraulic control unit, and

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

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

4.4.1.1.2 Establish a baseline APRM and LPRM* neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

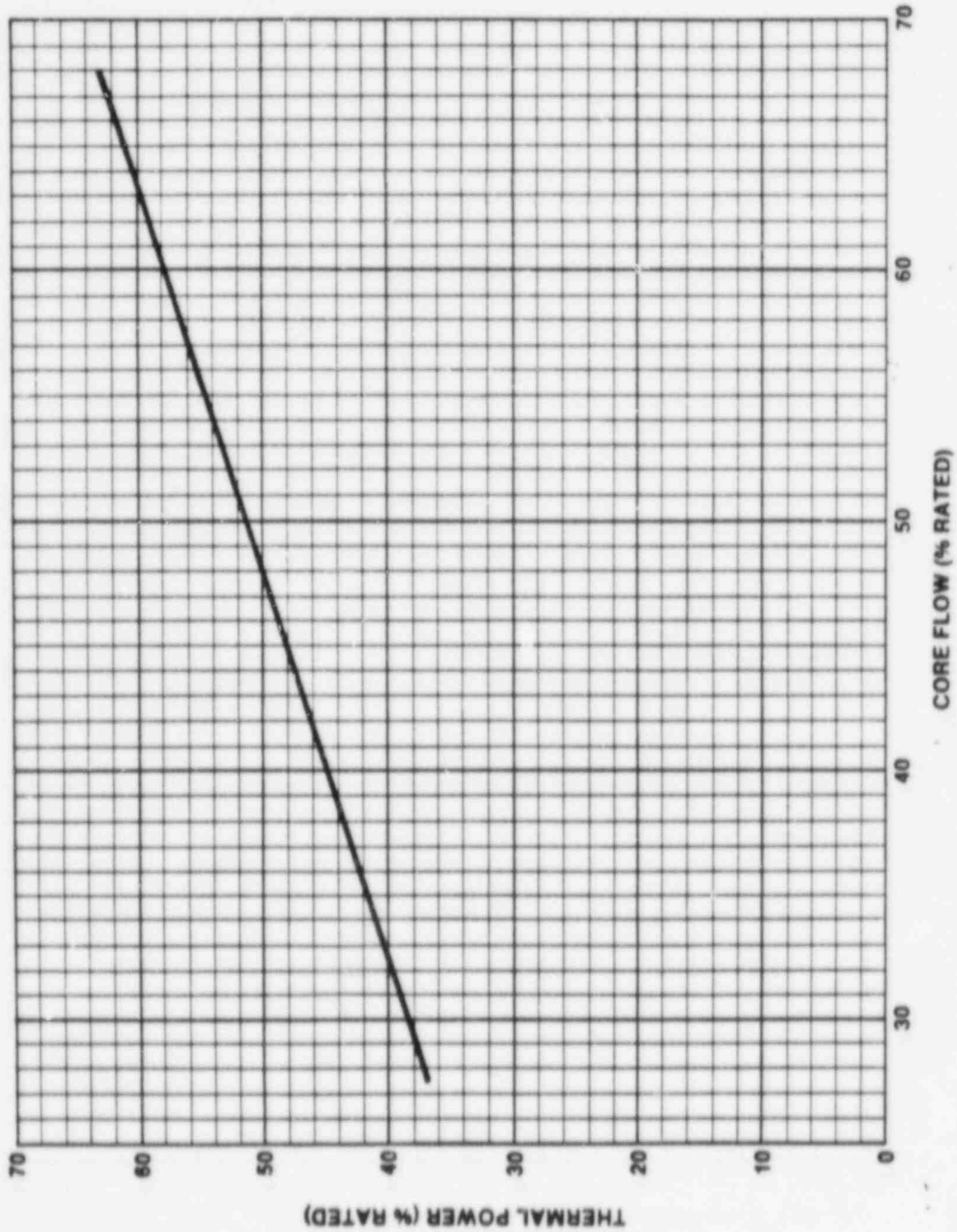

FIGURE 3.4.1.1-1
THERMAL POWER VERSUS CORE FLOW

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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 determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.

a. The indicated recirculation loop flow differs by more than 10% from the established* flow control 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 indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established* patterns by more than 10%.

*To be determined during the startup test program.

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RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare the recirculation loop with the lower flow not in operation and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

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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 pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F,* and:

a. When both loops have been idle, unless 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, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F 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 15 minutes prior to startup of an idle recirculation loop.

*Below 25 psig, this temperature differential is not applicable.

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REACTOR COOLANT SYSTEM

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3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 5 of the following valves and the relief valve function of at least 4 additional valves of the following valves, other than those satisfying the safety valve function requirement, shall be OPERABLE with the specified lift settings:

Number of Valves	Function	Setpoint* (psig)
7	Safety	1165 ± 1%
5	Safety	1180 ± 1%
4	Safety	1190 ± 1%
1	Relief	1103 ± 15 psig
8	Relief	1113 ± 15 psig
7	Relief	1123 ± 15 psig

The acoustic monitor for each OPERABLE valve shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the safety and/or relief 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.

b. With one or more safety/relief valves stuck open, close the stuck open safety/relief valve(s); if suppression pool average water temperature is 105°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) 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.

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

4.4.2.1.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE by performance of a:

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

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

4.4.2.1.2 The relief valve function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit setpoint, at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

*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 test.

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SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint* (psig) ± 15 psi Open	Close
1B21*RVF051D	1033	926
1B21*RVF051C	1073	936
1B21*RVF051B	1113	946
1B21*RVF051G	1113	946
1B21*RVF047F	1113	946

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and the low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. With either low-low set function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit setpoint, at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

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

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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 drywell atmosphere particulate radioactivity monitoring system,

b. The drywell and pedestal floor sump drain flow monitoring systems,

c. Either the drywell air coolers condensate flow rate monitoring system or the drywell atmosphere gaseous 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 drywell 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. Drywell atmosphere particulate and gaseous monitoring systems-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. The sump drain flow monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Drywell 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.

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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.

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

d. 1 gpm leakage at a reactor coolant system pressure of 1025 ± 15 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

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 pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual, deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

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SURVEILLANCE REQUIREMENTS

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

a. Monitoring the drywell atmospheric particulate radioactivity at least once per 12 hours,

b. Monitoring the sump flow rates at least once per 12 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

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

b. CHANNEL CALIBRATION at least once per 18 months.

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TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NUMBER	FUNCTION
1. LPCS	1E21*AOVF006	LPCS Injection
	1E21*MOVF005	LPCS Injection
2. HPCS	1E22*AOVF005	HPCS Injection
	1E22*MOVF004	HPCS Injection
3. RCIC	1E51*AOVF065	RCIC Head Spray
	1E51*MOVF013	RCIC Head Spray
4. RHR	1E12*MOVF023	RHR Head Spray
	1E12*AOVF041A	LPCI A Injection
	1E12*MOVF042A	LPCI A Injection
	1E12*AOVF041B	LPCI B Injection
	1E12*MOVF042B	LPCI B Injection
	1E12*AOVF041C	LPCI C Injection
	1E12*MOVF042C	LPCI C Injection
	1E12*MOVF009	Shutdown Cooling A & B Suction
	1E12*MOVF008	Shutdown Cooling A & B Suction

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

INSTRUMENT NUMBER	FUNCTION	ALARM SETPOINT
1E21*PTN054	LPCS Pump Discharge Pressure High	≤ 580 psig
1E22*PTN052	HPCS Pump Suction Pressure High	≤ 80 psig
1E51*PTN052	RCIC Pump Suction Pressure High	≤ 80 psig
1E12*PTN053A	RHR A Pump Discharge Pressure High	≤ 480 psig
1E12*PTN053B	RHR B Pump Discharge Pressure High	≤ 480 psig
1E12*PTN053C	RHR C Pump Discharge Pressure High	≤ 480 psig
1E12*PTN057	RHR Pump Shutdown Cooling Suction Pressure High	≤ 180 psig

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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, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and 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 and chloride concentration exceeding the limits specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

   3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration 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 CONDITIONS 2 and 3, with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

   1. With the:

      a) 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, or

      b) Chloride concentration exceeding the limit specified in Table 3.4.4-1, restore the chloride concentration to within the limit within 24 hours, or

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LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Prior to proceeding to OPERATIONAL CONDITON 3, determine that the structural integrity of the reactor coolant system remains acceptable for continued operation.

2. The provisions of Specification 3.0.3 are not applicable.

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 for:

1. Chlorides at least once per:

a) 72 hours, and

b) 8 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

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 the limit in Table 3.4.4-1.

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SURVEILLANCE REQUIREMENTS (Continued)

c. Continuously recording the conductivity of the reactor coolant or, when the continuous recording conductivity monitor is inoperable for up to 31 days, 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 CHECK 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 the limit in Table 3.4.4-1.

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TABLE 3.4.4-1

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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

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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 does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six-month period, prepare and submit within 30 days a Special Report to the Commission, pursuant to Specification 6.9.2, 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 for more than 800 hours cumulative operating time in a consecutive 12-month period, 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 steam line isolation valves closed, within 12 hours.

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, 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 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A Special Report shall be prepared and submitted to the Commission within 30 days, pursuant to Specification 6.9.2. This report shall contain the results of the specific activity analyses,

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the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131, and the following Additional Information.

c. In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in one hour*, or

2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

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 Special Report shall be prepared and submitted to the Commission at least once per 92 days, pursuant to Specification 6.9.2. This report shall contain, for each occurrence, the results of the specific activity analysis and the following Additional Information.

Additional Information

1. Reactor power history starting 48 hours prior to:

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

   b) The THERMAL POWER and/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

   b) The THERMAL POWER and/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

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

*Not applicable during the startup test program.

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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 Beta and Gamma 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 Ē Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	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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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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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,

c. A maximum temperature change of 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. 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 structural integrity of the reactor coolant system; and determine that the reactor coolant system remains acceptable for continued operations. Otherwise, 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, the reactor coolant system temperature and pressure shall be determined, at least once per 30 minutes, to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A and A', B and B', or C and C', as applicable.

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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 and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations 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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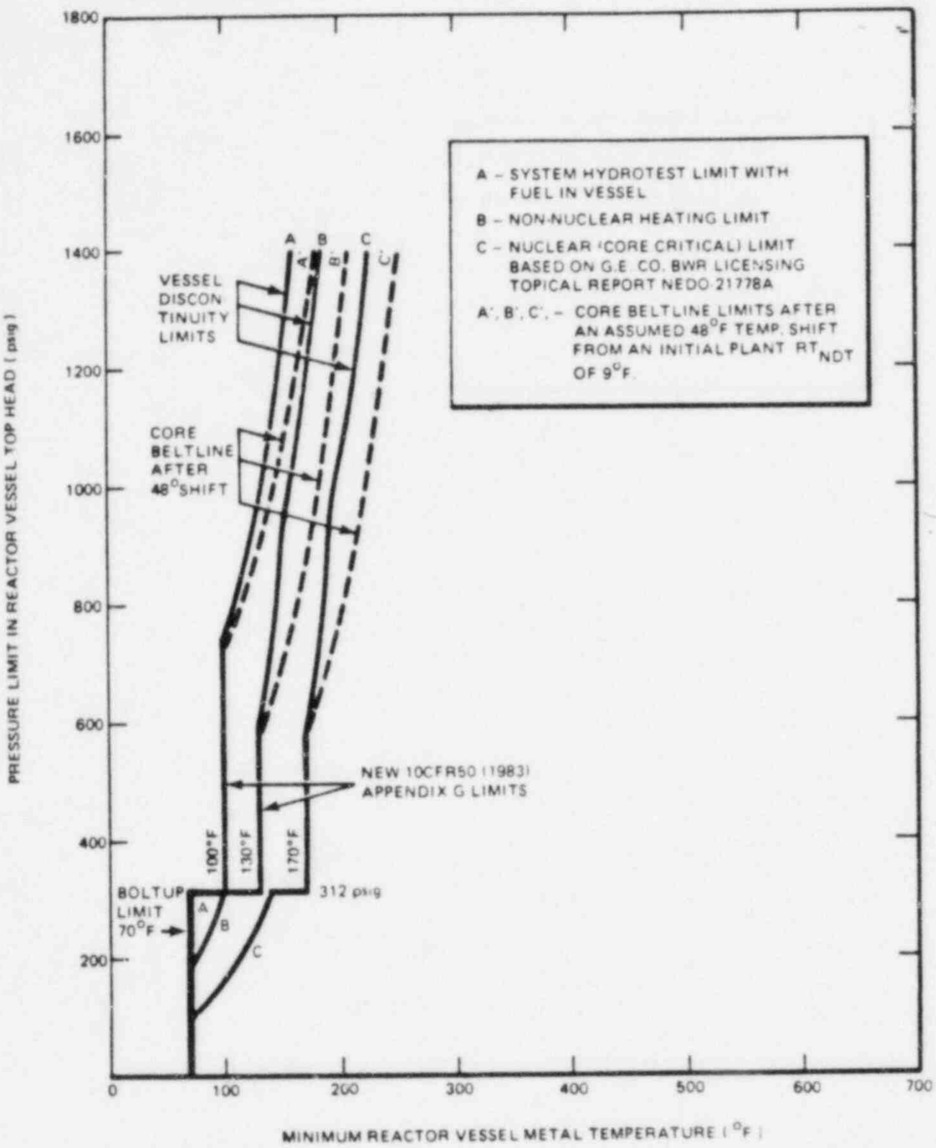

NOTE: CURVES A, B & C ARE PREDICTED TO APPLY AS THE LIMITS FOR 11 YEARS (8.8 EFPY) OF OPERATION.

FIGURE 3.4.6.1-1
MINIMUM TEMPERATURE REQUIRED VS REACTOR PRESSURE

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TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ ¼T	WITHDRAWAL TIME (EFPY)
1	3°	0.86	6
2	177°	0.86	15
3	183°	0.86	Standby

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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 CONDITIONS 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 At least once per 12 hours, the reactor steam dome pressure shall be verified to be less than 1045 psig.

*Not applicable during anticipated transients.

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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 seconds and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and, within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 At least once per 18 months, 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 not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component 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 not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component prior to increasing the Reactor Coolant System temperature above 200°F.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5.

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3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR Cut-in permissive setpoint.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.**

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, immediately initiate corrective action to return at least one RHR shutdown cooling loop or recirculation pump to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least once per 12 hours, at least one shutdown cooling mode loop of the residual heat removal system, one recirculation pump or alternate method shall be determined to be in operation and circulating reactor coolant.

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

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

**Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within one hour and at least once per 24 hours thereafter demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop or recirculation pump in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least once per 12 hours, at least one shutdown cooling mode loop of the residual heat removal system, one recirculation pump or alternate method shall be determined to be in operation and circulating reactor coolant.

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

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

##The shutdown cooling mode loop may be removed from operation during hydrostatic testing.

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3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions I, II, and III shall be OPERABLE with:

a. ECCS division I consisting of:

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

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

3. 7 OPERABLE ADS valves.

b. ECCS division II consisting of:

1. The 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 pool and transferring the water to the reactor vessel.

2. 7 OPERABLE ADS valves.

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

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 100 psig.

#See Special Test Exception 3.10.5.

##One LPCI subsystem of the RHR system may be aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR Cut-in permissive setpoint.

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ACTION:

a. For ECCS division I, provided that ECCS divisions II and III are OPERABLE:

    1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.

    2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

    3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

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

b. For ECCS division II, provided that ECCS divisions I and III are OPERABLE:

    1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

    2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

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

c. For ECCS division III, provided that ECCS divisions I and II and the RCIC system are OPERABLE:

    1. With ECCS division III inoperable, declare the 1C diesel inoperable and restore the inoperable division to OPERABLE status within 14 days.

    2. Otherwise, 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 as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For ECCS divisions I and II, provided that ECCS division III is OPERABLE:

1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

2. With the LPCS system inoperable and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCS system or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

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

e. For ECCS divisions I and II, provided that ECCS division III is OPERABLE and divisions I and II are otherwise OPERABLE:

1. With one of the above required ADS valves inoperable, 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 $\leq$ 100 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to $\leq$ 100 psig within the next 24 hours.

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

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

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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4.5.1 ECCS division I, II and III shall be demonstrated OPERABLE by:

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

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   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. Verifying that, when tested pursuant to Specification 4.0.5, each:

   1. LPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 281 psid.

   2. LPCI pump develops a flow of at least 5050 gpm with a pump differential pressure greater than or equal to 100 psid.

   3. HPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 399 psid.

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

d. At least once per 18 months, for the HPCS system, verifying that the suction 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, and verifying that the HPCS system will automatically restart on Reactor Vessel Water Level - Low Low Level 2.

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SURVEILLANCE REQUIREMENTS (Continued)

e. 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:

   a) The control valve or bypass valve position responds accordingly, or

   b) There is a corresponding change in the measured steam flow, or

   c) The acoustic monitoring system indicates the valve is open.

*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 test.

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3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following shall be OPERABLE:

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

b. Low-pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

c. Low-pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

d. Low-pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

e. The high-pressure core spray (HPCS) system with a flow path capable of taking suction from the condensate storage tank or suppression pool, as applicable, when these sources of water are OPERABLE per Specification 3.5.3.b, and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/ system to OPERABLE status within 4 hours or establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING within the next 8 hours.

*The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the upper containment fuel pool gate is opened, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

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SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

4.5.2.2 At least once per 12 hours the HPCS system shall be determined OPERABLE by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.e.

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3/4.5.3 SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3 with a contained water volume of at least 137,571 ft^3, equivalent to a level of 19'6".

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least 94,000 ft^3, equivalent to a level of 13'3", except that the suppression pool level may be less than the limit or may be drained provided that:

1. No operations are performed that have a potential for draining the reactor vessel,

2. The reactor mode switch is locked in the Shutdown or Refuel position,

3. The condensate storage tank contains at least 125,000 available gallons of water, equivalent to a level of 11'1", and

4. 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 pool 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 or 5* with the suppression pool 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. Establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING within 8 hours.

---

*The suppression pool is not required to be OPERABLE in OPERATIONAL CONDITION 5 provided that the reactor vessel head is removed, the cavity is flooded, the upper containment fuel pool gate is open, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

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SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

a. 19'6", at least once per 24 hours, in OPERATIONAL CONDITION 1, 2 and 3.

b. 13'3", at least once per 12 hours, in OPERATIONAL CONDITION 4 and 5.

4.5.3.2 With the suppression pool level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

*The suppression pool is not required to be OPERABLE in OPERATIONAL CONDITION 5 provided that the reactor vessel head is removed, the cavity is flooded, the upper containment fuel pool gate is open, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

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3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY - OPERATING shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY - OPERATING, restore PRIMARY CONTAINMENT INTEGRITY - OPERATING 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 - OPERATING shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 7.6 psig, and verifying that, when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.3.d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 La.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE primary 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 Specification 3.6.4.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.4.

d. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, steam tunnel or drywell, 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 INTEGRITY - FUEL HANDLING

LIMITING CONDITION FOR OPERATION

3.6.1.2 PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING shall be maintained.

APPLICABILITY: Operational Condition*

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING, suspend handling of irradiated fuel in the primary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.6.1.2 PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING shall be demonstrated:

a. Within 24 hours prior to entering and at least once per 24 hours during Operational Condition* by verifying that all primary containment penetrations required to be closed during accident conditions are closed by hatches, valves, blind flanges, or deactivated automatic valves secured in position.

b. By verifying each containment air lock is in compliance with the requirements of Specification 3.6.1.4.

*When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

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PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.3 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to La, 0.26 percent by weight of the primary containment air per 24 hours at Pa, 7.6 psig.

b. A combined leakage rate of less than 0.60 La for all penetrations and all valves subject to Type B and C tests when pressurized in accordance with Table 3.6.4-1 of Specification 3.6.4.

c. A leakage rate of less than 150 scfh for the valves served by each Division of MS-PLCS and a leakage rate of less than 340 scfh for each of the valve groups identified below when tested in accordance with the surveillance requirements of 4.6.1.3.f.

1. Division I MS-PLCS Valves and Division I PVLCS Valves
2. Division II MS-PLCS Valves and Division II PVLCS Valves
3. Division I MS-PLCS Valves and all first outboard PVLCS Valves

d. A combined leakage rate of less than or equal to 13,500 cc/hr for all penetrations shown in Table 3.6.1.3-1 as annulus bypass leakage paths when pressurized to Pa, 7.6 psig.

e. A combined leakage rate of less than or equal to 170,000 cc/hr, for all valves shown in Table 3.6.4-1 to be secondary containment bypass leakage paths and equipped with PVLCS, when pressurized to Pa, 7.6 psig.

f. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, when tested at 1.1 Pa, 8.36 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY-OPERATING is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate equaling or exceeding 0.75 La, or

b. The measured combined leakage rate, for all penetrations and all valves subject to Type B and C tests, exceeding 0.60 La, or

c. The measured leakage rate greater than or equal to 150 scfh for the valves served by each Division of MS-PLCS or the measured leakage rate greater than or equal to 340 scfh for each valve grouping identified in 3.6.1.3.c.1, 3.6.1.3.c.2 or 3.6.1.3.c.3, or

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. The combined leakage rate, for all penetrations shown in Table 3.6.1.3-1 as annulus bypass leakage paths, exceeding 13,500 cc/hr, or

e. The combined leakage rate, for all valves shown in Table 3.6.4-1 to be secondary containment bypass leakage paths and equipped with PVLCS, exceeding 170,000 cc/hr, or

f. The measured combined leakage rate, for all containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) to less than 0.75 La as applicable, and

b. The combined leakage rate, for all penetrations and all valves subject to Type B and C tests, to less than 0.60 La, and

c. The measured leakage rate to less than 150 scfh for the valves served by each Division of MS-PLCS and the measured leakage rate to less than 340 scfh for each of the valve groupings identified in 3.6.1.3.c.1, 3.6.1.3.c.2, and 3.6.1.3.c.3, and

d. The combined leakage rate, for all penetrations shown in Table 3.6.1.3-1 as annulus bypass leakage paths, to less than or equal to 13,500 cc/hr, and

e. The combined leakage rate, for all valves shown in Table 3.6.4-1 to be secondary containment bypass leakage paths and equipped with PVLCS, to less than or equal to 170,000 cc/hr, and

f. The combined leakage rate, for all containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary 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 Pa, 7.6 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 0.75 La, 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 0.75 La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La, 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 test data and the Type A test data is within 0.25 La. The formula to be used is: [Lo + Lam - 0.25 La] $\leq$ Lc $\leq$ [Lo + Lam + 0.25 La] where Lc = supplemental test results; Lo = superimposed leakage; Lam = measured Type A leakage.

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 primary containment or bled from the primary containment during the supplemental test, to be between 0.75 La and 1.25 La.

d. Type B and C tests shall be conducted with gas at Pa, 7.6 psig*, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam positive leakage control system (MS-PLCS) valves and PVLCS valves,

3. Penetrations using continuous leakage monitoring systems,

4. Primary containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, and

5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.4.

f. Total sealing air leakage into the primary containment, at a test pressure of 11.5 psid for MS-PLCS valves and 33 psid for penetration leakage control system sealed valves, shall be determined by test at least once per 18 months. This leakage may be excluded when determining the combined leakage rate, 0.6 La.

*Unless a hydrostatic test is required per Table 3.6.4-1.

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SURVEILLANCE REQUIREMENTS (Continued)

g. Type B tests for electrical penetrations employing a continuous leakage monitoring system shall be conducted at Pa, 7.6 psig, at intervals no greater than once per 3 years.

h. Leakage from isolation valves that are sealed with the PVLCS shall be tested once per 24 months with the valves pressurized to at least Pa, 7.6 psig. This leakage may be excluded when determining the combined leakage rate, 0.6 La.

i. Primary containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment shall be leak tested at least once per 18 months.

j. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.9.3.

k. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.3.a, 4.6.1.3.d, 4.6.1.3.g, and 4.6.1.3.h.

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TABLE 3.6.1.3-1

ANNULUS BYPASS LEAKAGE PATHS

1. LEAKAGE PATHS TO THE FUEL BUILDING (6750 cc/hr limit)

PENETRATION	
Containment air lock	1JRB*DRA2

2. LEAKAGE PATHS TO THE AUXILIARY BUILDING (6750 cc/hr limit)

PENETRATION	VALVE NO. (DIV. I)	VALVE NO. (DIV. II)
1KJB*Z31	1HVR*AOV165	1HVR*AOV123
1KJB*605E	1CMS*SOV31A	1CMS*SOV35C
1KJB*605F	1CMS*SOV31C	1CMS*SOV35A
1KJB*601B	1SSR*SOV131	1SSR*SOV130
Containment air lock	1JRB*DRA1	
CRD removal hatch	--	

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PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.4 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed, except at least one air lock door shall be closed when the air lock is being used for normal transit entry and exit through the containment, and

b. An overall air lock leakage rate in compliance with the limits of Specification 3.6.1.3.d when pressurized to Pa, 7.6 psig, and

c. The inflatable seal system air flask pressure $\geq$ 90 psig.

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

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.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, in OPERATIONAL CONDITIONS 1, 2, or 3 be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. Otherwise, in Operational Condition #, suspend all operations involving handling of irradiated fuel in the containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

5. The provisions of Specification 3.0.4 are not applicable.

b. With a 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:

1. In OPERATIONAL CONDITIONS 1, 2, or 3, 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.
#When irradiated fuel is being handled in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

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LIMITING CONDITION FOR OPERATION (Continued)

2. In Operational Condition #, suspend all operations involving handling of irradiated fuel in the containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

c. With one primary containment air lock door inflatable seal system air flask pressure instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify, at least once per 12 hours, air flask pressures to be $\geq$ 90 psig.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each primary containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except at least once per 72 hours when the air lock is being used for multiple entries, by verifying seal leakage rate is in compliance with the limits in Specification 3.6.1.3.d when the gap between the door seals is pressurized to Pa, 7.6 psig.

b. By conducting an overall air lock leakage test at Pa, 7.6 psig, and verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months,*

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY - OPERATING when the air lock has been used or when maintenance has been performed on the air lock that could affect the air lock sealing capability.

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

d. By verifying the door inflatable seal system OPERABLE by:

1. At least once per 7 days, verifying seal air flask pressure to be greater than or equal to 90 psig.

2. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 1.28 psig from 90 psig within 24 hours.

*The provisions of Specification 4.0.2 are not applicable.

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MAIN STEAM-POSITIVE LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.5 Two independent main steam positive leakage control system (MS-PLCS) divisions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MS-PLCS division inoperable, restore the inoperable division 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.

SURVEILLANCE REQUIREMENTS

4.6.1.5 Each MS-PLCS division shall be demonstrated OPERABLE:

a. By performing Surveillance Requirement 4.6.1.10.a.

b. At least once per 31 days by verifying compressor OPERABILITY by operating the compressor loaded for at least 15 minutes.

c. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor-operated valve, including the Main Steam Shutoff Valves (MSSV), through at least one complete cycle of full travel.

d. At least once per 18 months by performance of a functional test, which includes simulated actuation of the division hroughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that 8.5 ± 3 psid sealing pressure is established in each steam line.

e. By verifying the operating instrumentation to be OPERABLE by performance of a:

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

2. CHANNEL CALIBRATION at least once per 18 months.

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PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

4.6.1.6 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined, during the shutdown for each Type A primary containment leakage rate test, by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A primary containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

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PRIMARY CONTAINMENT INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Primary containment internal pressure shall be maintained between -0.3 and +0.3 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the primary containment internal pressure outside of the specified limits, restore the pressure 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.

SURVEILLANCE REQUIREMENTS

4.6.1.7 At least once per 12 hours, the primary containment internal pressure shall be determined to be within the limits.

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PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.8 Primary containment average air temperature shall not exceed 90°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the primary containment average air temperature greater than 90°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

4.6.1.8 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined, at least once per 24 hours, to be within the limit:

	ELEVATION	AZIMUTH*
a.	~167'	~37°, ~72°, ~108°
b.	~122'	~170°
c.	~119'	~15°, ~66°, ~117°, ~219°, ~270°, ~322°

*At least one reading from each elevation is required for an average calculation. However, if all instrumentation is OPERABLE, all readings should be used in the calculation.

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PRIMARY CONTAINMENT PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 The primary containment purge 36-inch supply and exhaust isolation valves shall be OPERABLE and closed except:

a. Each 36-inch purge valve may be open for purge system operation, with such operation limited to 1000 hours* per 365 days, for reducing airborne activity and for pressure control, and

b. If the SGTS is in the purge flow path, both trains of the SGTS must be OPERABLE, but only one train of SGTS may be operating in the purge flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With a 36-inch primary containment purge supply or exhaust isolation valve open for more than 1000 hours* per 365 days, close and/or seal the 36-inch valve or otherwise isolate the penetration within 4 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 SGTS trains in operation in the purge flow path or without both SGTS OPERABLE with one SGTS in the purge flow path, discontinue 36-inch purge system operation and close the open 36-inch valve(s) or otherwise isolate the penetration(s) 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 a primary containment purge supply or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.9.3, restore the inoperable valve(s) 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.

SURVEILLANCE REQUIREMENTS

4.6.1.9.1 Each 36 inch primary containment purge supply and exhaust isolation valve shall be verified to be closed at least once per 31 days.

4.6.1.9.2 The cumulative time that the 36-inch primary containment purge supply and/or exhaust isolation valves have been open during the past 365 days shall be determined at least once per 7 days.

*Limit of 2000 hours per 365 days from initial fuel loading until 3 months after the first refueling outage.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.9.3 At least once per 92 days, each 36-inch primary containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is in accordance with the limits in Specification 3.6.1.3.d when pressurized to Pa, 7.6 psig, for each purge supply valve, and is less than or equal to 0.01 La when pressurized to Pa for each purge exhaust valve.

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PENETRATION VALVE LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.10 Two independent penetration valve leakage control system (PVLCS) divisions shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one PVLCS division inoperable, restore the inoperable division 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

4.6.1.10 Each PVLCS division shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying division PVLCS accumulator pressure greater than or equal to 101 psig.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor-operated 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 system throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that a sealing pressure greater than or equal to 22 psig is established in each sealing valve.

d. By verifying the operating instrumentation to be OPERABLE by performance of a:

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

   2. CHANNEL CALIBRATION at least once per 18 months.

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3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

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

ACTION:

Without DRYWELL INTEGRITY, restore DRYWELL 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

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all drywell penetrations**, not capable of being closed by OPERABLE drywell 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 Specification 3.6.4.

b. By verifying the drywell air lock is in compliance with the requirements of Specification 3.6.2.3.

c. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

d. By verifying the drywell bypass leakage is in compliance with the requirements of Specification 3.6.2.2.

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the drywell or primary 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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SURVEILLANCE REQUIREMENTS (Continued)

e. By verifying the personnel door inflatable seal system OPERABLE by:

   1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 75 psig.

   2. At least once per 18 months conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 0.67 psig from 75 psig within 24 hours.

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LIMITING CONDITION FOR OPERATION

3.6.2.2 Drywell bypass leakage shall be less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 1.0 ft^2.

APPLICABILITY: When DRYWELL INTEGRITY is required per Specification 3.6.2.1.

ACTION:

With the drywell bypass leakage greater than 10% of the acceptable $A/\sqrt{k}$ design value of 1.0 ft^2, restore the drywell bypass leakage to within the limit prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.2 At least once per 18 months, the drywell bypass leakage rate test shall be conducted at an initial differential pressure of 3.0 psid and the $A/\sqrt{k}$ shall be calculated from the measured leakage. One drywell airlock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other leakage rate test.

a. If any drywell bypass leakage test fails to meet the specified limit, the schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the limit, a test shall be performed at least every 9 months until two consecutive tests meet the limit, at which time the 18 month test schedule may be resumed.

b. The provisions of Specification 4.0.2 are not applicable.

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DRYWELL AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.2.3 The drywell air lock shall be OPERABLE with:

a. Both doors closed except that, when the air lock is being used for normal transit entry and exit through the drywell, at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 11.85 scf per hour at 3.0 psid, and

c. The inflatable seal system air flask pressure $\geq$ 75 psig.

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

ACTION:

a. With one drywell 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 provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. The provisions of Specification 3.0.4 are not applicable.

b. With the drywell air lock inoperable, except as a result of an inoperable air lock door, mainta n 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.

c. With one inoperable drywell air lock door inflatable seal system air flask pressure instrumentation channel, restore the inoperable channel to OPERABLE status within 7 days or verify air flask pressure to be $\geq$ 75 psig at least once per 12 hours.

*See Special Test Exception 3.10.1.

4.6.2.3 The drywell air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 4.05 scf per hour when the gap between the door seals is pressurized to 3.0 psid.

b. By conducting an overall air lock leakage test at 3.0 psid and verifying that the overall air lock leakage rate is within its limit:

   1. Each cold shutdown if not performed within the previous 6 months#.

   2. Prior to establishing DRYWELL INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.

c. By verifying, prior to drywell entry if not performed within the past 18 months, that only one door in the air lock can be opened at a time.

d. By verifying the door inflatable seal system OPERABLE by:

   1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 75 psig.

   2. At least once per 18 months conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 0.67 psig from 75 psig within 24 hours.

#At least once per 18 months, the air lock will be pressurized to 19.2 psid prior to conducting the overall air lock test.

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DRYWELL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.4 The structural integrity of the drywell shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.2.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the drywell 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

4.6.2.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the drywell 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.

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DRYWELL INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.2.5 Drywell-to-primary containment differential pressure shall be maintained between - 0.3 and + 1.2 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell-to-primary containment differential pressure outside of the specified limits, restore the differential pressure 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.

SURVEILLANCE REQUIREMENTS

4.6.2.5 At least once per 12 hours, the drywell-to-primary containment differential pressure shall be determined to be within the limits.

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CONTAINMENT SYSTEMS

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DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.2.6 Drywell average air temperature shall not exceed 140°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 140°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

4.6.2.6 The drywell average air temperature shall be the arithmetical 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.	~145'	$20° \leq A \leq 60°$
b.	~145'	$100° \leq A \leq 150°$
c.	~145'	$190° \leq A \leq 265°$
d.	~145'	$290° \leq A \leq 330°$

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DRYWELL VENT AND PURGE

LIMITING CONDITION FOR OPERATION

3.6.2.7 The drywell vent and purge system supply and exhaust valves shall be sealed closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the drywell vent and purge system supply or exhaust valves open in OPERATIONAL CONDITIONS 1, 2, or 3, immediately close the drywell vent and purge system valves or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.7 At least once per 31 days, verify the drywell vent and purge system supply and exhaust valves to be sealed closed.

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3/4.6.3 DEPRESSURIZATON SYSTEMS

SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.6.3.1 The suppression pool shall be OPERABLE with the pool water:

a. Volume between 137,571 ft^3 and 141,036 ft^3, equivalent to a level between 19'6" and 20'0" and a

b. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

1. 105°F during testing which adds heat to the suppression pool.

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.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression pool 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 pool average water temperature greater than 95°F, restore the average 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, except, as permitted above:

1. With the suppression pool average water temperature greater than 105°F during testing which adds heat to the suppression pool, stop all testing which adds heat to the suppression pool and restore the average 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.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. With the suppression pool average water temperature greater than:

   a) 95°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

   b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression pool average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With only one suppression pool water level indicator OPERABLE and/or with fewer than eight suppression pool water temperature indicators, one in each of eight locations, OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify, at least once per 12 hours, suppression pool water level and/or temperature to be within the limits.

d. With no suppression pool water level indicators OPERABLE and/or with fewer than seven suppression pool water temperature indicators, covering at least seven locations, OPERABLE, restore at least one water level indicator and at least six water temperature indicators 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

4.6.3.1 The suppression pool shall be demonstrated OPERABLE:

a. By verifying, at least once per 24 hours, the suppression pool water volume to be within the limits.

b. At least once per 24 hours, in OPERATIONAL CONDITION 1 or 2, by verifying the suppression pool average water temperature to be less than or equal to 95°F, except:

   1. At least once per 5 minutes, during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature less than or equal to 105°F.

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SURVEILLANCE REQUIREMENTS (Continued)

2. At least once per hour, when suppression pool average water temperature is greater than or equal to 95°F, by verifying suppression pool average water temperature to be less than or equal to 110°F and THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER.

3. At least once per 30 minutes, following a scram, with suppression pool average water temperature greater than or equal to 95°F, by verifying suppression pool average water temperature less than or equal to 120°F.

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PRIMARY CONTAINMENT UNIT COOLERS

LIMITING CONDITION FOR OPERATION

3.6.3.2 Both primary containment unit coolers shall be OPERABLE and capable of rejecting heat to the Standby Service Water System.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one of the primary containment unit coolers inoperable, restore the inoperable unit cooler 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 primary containment unit coolers inoperable, restore at least one unit cooler 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

4.6.3.2 Both primary containment unit coolers shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each pressure relief and backdraft damper, 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 each of the required unit coolers develops a flow of at least 50,000 cfm on recirculation flow through the unit cooler.

c. At least once per 18 months by performance of a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each pressure relief and backdraft damper in the flow path actuates to its correct position.

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SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.3.3 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. An OPERABLE flow path capable of recirculating water from the suppression pool through an RHRSW 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 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.3 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 5050 gpm on recirculation flow through the RHR heat exchangers to the suppression pool when tested pursuant to Specification 4.0.5.

*Whenever both RHR loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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3/4.6.4 PRIMARY CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 The primary containment and drywell isolation valves in Table 3.6.4-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.4-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more of the primary containment or drywell isolation valves shown in Table 3.6.4-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and, within 4 hours, either:

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

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

d. The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION b. or c. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each isolation valve shown in Table 3.6.4-1 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.

4.6.4.2 Each automatic isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that, on an isolation test signal, each automatic isolation valve actuates to its isolation position.

4.6.4.3 The isolation time of each power operated or automatic valve shown in Table 3.6.4-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
a. Automatic Isolation Valves					
1. Primary Containment(a)					
MSIV	1B21*AOVF022A(b)(g)	1KJB*Z1A	6	5	No
MSIV	1B21*AOVF022B(b)(g)	1KJB*Z1B	6	5	No
MSIV	1B21*AOVF022C(b)(g)	1KJB*Z1C	6	5	No
MSIV	1B21*AOVF022D(b)(g)	1KJB*Z1D	6	5	No
MSIV	1B21*AOVF028A(g)	1KJB*Z1A	6	5	No
MSIV	1B21*AOVF028B(g)	1KJB*Z1B	6	5	No
MSIV	1B21*AOVF028C(g)	1KJB*Z1C	6	5	No
MSIV	1B21*AOVF028D(g)	1KJB*Z1D	6	5	No
Turbine Plant Misc. Drains	1B21*MOVF067A(g)	1KJB*Z1A	6	17.8	No
Turbine Plant Misc. Drains	1B21*MOVF067B(g)	1KJB*Z1B	6	16.1	No
Turbine Plant Misc. Drains	1B21*MOVF067C(g)	1KJB*Z1C	6	15.9	No
Turbine Plant Misc. Drains	1B21*MOVF067D(g)	1KJB*Z1D	6	19.8	No
Turbine Plant Misc. Drains	1B21*MOVF016(b)(g)	1KJB*Z2	6	16.5	No
Turbine Plant Misc. Drains	1B21*MOVF019(g)	1KJB*Z2	6	17.6	No
RHR Return to FW	1E12*MOVF053A	1KJB*Z3A	5	18.7	No
RHR Return to FW	1E12*MOVF053B	1KJB*Z3B	5	18.7	No
RHR/RCIC Head Supply	1E12*MOVF023(b)	1KJB*Z19, 1DRB*Z13	5	36.3	No
RHR Shutdown Cooling Supply	1E12*MOVF008	1KJB*Z20	5	29.7	No
RHR Shutdown Cooling Supply	1E12*MOVF009(b)	1KJB*Z20	5	25.3	No
LPCI A to Reactor	1E12*MOVF037A	1KJB*Z21A	14	73.7	No
LPCI B to Reactor	1E12*MOVF037B	1KJB*Z21B	14	74.8	No
MS-PLCS Line	1E33*MOVF008(d)	1KJB*Z1A,B,C,D	4	14.5	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
a. Automatic Isolation Valves					
1. Primary Containment[a] (Continued)					
RWCU Disch. to Condenser	1G33*MOVF028	1KJB*Z4	15	20.9	Yes
RWCU Return to FW	1G33*MOVF040	1KJB*Z6	15	24.2	No
RWCU Pump Suction	1G33*MOVF001[b]	1KJB*Z7	16	19.8	No
RWCU Pump Disch.	1G33*MOVF053	1KJB*Z129	15	5.5	No
RWCU Disch. to Condenser	1G33*MOVF034	1KJB*Z4	15	20.9	Yes
RWCU Return to FW	1G33*MOVF039	1KJB*Z6	15	24.2	No
RWCU Pump Suction	1G33*MOVF004	1KJB*Z7	7	6.6	No
RWCU Pump Disch.	1G33*MOVF054	1KJB*Z129	15	5.5	No
RWCU Backwash Disch.	1WCS*MOV178	1KJB*Z5	1	12.1	Yes
RWCU Backwash Disch.	1WCS*MOV172	1KJB*Z5	1	12.6	Yes
HPCS Test Return-Supp. Pool	1E22*MOVF023(j)	1KJB*Z11	1	50	No
RHR A Return-Supp. Pool	1E12*MOVF024A(j)	1KJB*Z24A	10	63.8	No
RHR A Hx Dump-Supp. Pool	1E12*MOVF011A(j)	1KJB*Z24A	10	34.1	No
LPCS Test Return-Supp. Pool	1E21*MOVF012(j)	1KJB*Z24A	10	57.2	No
RHR B Return-Supp. Pool	1E12*MOVF024B(j)	1KJB*Z24B	10	63.8	No
RHR B Hx Dump-Supp. Pool	1E12*MOVF011B(j)	1KJB*Z24B	10	30.8	No
RHR C Return-Supp. Pool	1E12*MOVF021(j)	1KJB*Z24C	10	97.9	No
Fuel Pool Co+Cl Disch.	1SFC*MOV119	1KJB*Z26	1	68	No
Fuel Pool Co+Cl Suction	1SFC*MOV120	1KJB*Z27	1	62.7	No
Fuel Pool Co+Cl Suction	1SFC*MOV122	1KJB*Z27	1	63.8	No
Fuel Pool Purif. Suction	1SFC*MOV139	1KJB*Z28	1	39.6	No
Fuel Pool Purif. Suction	1SFC*MOV121	1KJB*Z28	1	39.6	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
a. Automatic Isolation Valves					
1. Primary Containment[(a)] (Continued)					
Floor Drain Disch.	1DFR*AOV102[(b)]	1KJB*Z35, 1DRB*Z36	1	N/A	No
Floor Drain Disch.	1DFR*AOV101[(b)]	1KJB*Z35, 1DRB*Z36	1	N/A	No
Equip. Drain Disch.	1DER*AOV127[(b)]	1KJB*Z38, 1DRB*Z39	1	N/A	No
Equip. Drain Disch.	1DER*AOV126[(b)]	1KJB*Z38, 1DRB*Z39	1	N/A	No
Fire Protection Hdr.	1FPW*MOV121	1KJB*Z41	1	34.1	Yes
Service Air Supply	1SAS*MOV102	1KJB*Z44	1	22.0	Yes
Instr. Air Supply	1IAS*MOV106	1KJB*Z46	1	18.7	Yes
RPCCW Supply	1CCP*MOV138	1KJB*Z48	1	22.0	No
RPCCW Return	1CCP*MOV158	1KJB*Z49	1	23.1	No
RPCCW Return	1CCP*MOV159	1KJB*Z49	1	24.2	No
Service Water Return	1SWP*MOV5A	1KJB*Z53A	1	50.6	No
Service Water Return	1SWP*MOV5B	1KJB*Z53B	1	53.9	No
Vent. Chilled Water Rtn.	1HVN*MOV102	1KJB*Z131	1	31.9	Yes
Vent. Chilled Water Rtn.	1HVN*MOV128	1KJB*Z131	1	28.6	Yes
Vent. Chilled Water Sup.	1HVN*MOV127	1KJB*Z132	1	27.5	Yes
Condensate	1CNS*MOV125	1KJB*Z134	1	22.0	Yes

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
a. Automatic Isolation Valves					
1. Primary Containment[(a)] (Continued)					
RHR & RCIC Steam Sup.	1E51*MOVF063[(b)]	1KJB*Z15	2	9.9	No
RHR & RCIC Steam Sup.	1E51*MOVF076[(b)]	1KJB*Z15	2	13.4	No
RHR & RCIC Steam Sup.	1E51*MOVF064	1KJB*Z15	2	9.9	No
RCIC Pump Suc.-Supp. Pool	1E51*MOVF031[(j)]	1KJB*Z16	2	30.5	No
RCIC Turbine Exh.-Supp. Pool	1E51*MOVF077	1KJB*Z17	3	14.2	No
RCIC Turbine Exh. Vac. Bkrs.	1E51*MOVF078	1KJB*Z18B,C	3	16.5	No
Cont./Drywell Purge Sup.	1HVR*AOV165	1KJB*Z31	8	3	No
Cont./Drywell Purge Sup.	1HVR*AOV123	1KJB*Z31	8	3	No
Cont./Drywell Purge Outlet	1HVR*AOV128	1KJB*Z33	8	3	No
Cont./Drywell Purge Outlet	1HVR*AOV166	1KJB*Z33	8	3	No
Post-Accident Samp. Sup.	1SSR*SOV130	1KJB*Z601B	1	3	No
Post-Accident Samp. Sup.	1SSR*SOV131	1KJB*Z601B	1	3	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
a. Automatic Isolation Valves					
2. Drywell[(k)]					
Cont./Drywell Purge Sup.	1HVR*AOV147	1DRB*Z32	1	3	No
RPCCW Supply	1CCP*MOV142	1DRB*Z50	1	30	No
RPCCW Return	1CCP*MOV144	1DRB*Z51	1	30	No
RPCCW Return	1CCP*MOV143	1DRB*Z51	1	30	No
Service Water Supply	1SWP*MOV4A	1DRB*Z54	1	52.8	No
Service Water Supply	1SWP*MOV4B	1DRB*Z54	1	51.7	No
Service Water Return	1SWP*MOV5A	1DRB*Z55	1	50.6	No
Service Water Return	1SWP*MOV5B	1DRB*Z55	1	53.9	No
Recirc. Flow Control	1RCS*MOV58A	1DRB*Z152	1	11.0	No
Recirc. Flow Control	1RCS*MOV59A	1DRB*Z153	1	10.6	No
Recirc. Flow Control	1RCS*MOV60A	1DRB*Z154	1	6.3	No
Recirc. Flow Control	1RCS*MOV61A	1DRB*Z155	1	8.6	No
Recirc. Flow Control	1RCS*MOV58B	1DRB*Z156	1	10.6	No
Recirc. Flow Control	1RCS*MOV59B	1DRB*Z157	1	10.8	No
Recirc. Flow Control	1RCS*MOV60B	1DRB*Z158	1	6.38	No
Recirc. Flow Control	1RCS*MOV61B	1DRB*Z159	1	8.9	No
Cont./Drywell Purge Sup.	1HVR*AOV125	1DRB*Z32	1	3	No
Cont./Drywell Purge Rtn.	1HVR*AOV126	1DRB*Z34	1	3	No
Cont./Drywell Purge Rtn.	1HVR*AOV148	1DRB*Z34	1	3	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
a. Automatic Isolation Valves					
2. Drywell(k) (Continued)					
Hydrogen Mixing Line	1CPM*MOV2A	1DRB*Z57A	1	33	No
Hydrogen Mixing Line	1CPM*MOV4A	1DRB*Z57A	1	33	No
Hydrogen Mixing Line	1CPM*MOV2B	1DRB*Z57B	1	33	No
Hydrogen Mixing Line	1CPM*MOV4B	1DRB*Z57B	1	33	No
Hydrogen Mixing Line	1CPM*MOV3A	1DRB*Z58A	1	33	No
Hydrogen Mixing Line	1CPM*MOV1A	1DRB*Z58A	1	33	No
Hydrogen Mixing Line	1CPM*MOV3B	1DRB*Z58B	1	33	No
Hydrogen Mixing Line	1CPM*MOV1B	1DRB*Z58B	1	33	No
Reactor Plant Sampling	1B33*AOVF019	1DRB*Z449	9	5	No
Reactor Plant Sampling	1B33*AOVF020	1DRB*Z449	9	5	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
b. Manual Isolation Valves			
1. Primary Containment(a)			
LPCI A to Reactor	1E12*F099A	1KJB*Z21A	No
LPCI B to Reactor	1E12*F099B	1KJB*Z21B	No
Reactor Plant Vent. ΔP Trans.	1HVR*V8	1KJB*Z602A	No
Reactor Plant Vent. ΔP Trans.	1HVR*V10	1KJB*Z602B	No
PVLCS Pressure Transmitter	1LSV*V64	1KJB*Z602D	No
Reactor Plant Vent. ΔP Trans.	1HVR*V12	1KJB*Z602F	No
Cont. Leakage Monitor Press.	1LMS*V14	1KJB*Z603A	No
Cont. Leakage Monitor Press.	1LMS*V12	1KJB*Z603A	No
Cont. Leakage Monitor Press.	1LMS*V7	1KJB*Z603C	No
Cont. Leakage Monitor Press.	1LMS*V16	1KJB*Z603C	No
Cont. Monitor Press. Sensing	1CMS*V2	1KJB*Z605A	No
Cont. Monitor Press. Sensing	1CMS*V3	1KJB*Z605B	No
Reactor Plant Vent. ΔP Trans.	1HVR*V14	1KJB*Z606A	No
Reactor Plant Vent. ΔP Trans.	1HVR*V16	1KJB*Z606B	No
Cont. Monitor Press. Sensing	1CMS*V16	1KJB*Z606C	No
Cont. Monitor Press. Sensing	1CMS*V15	1KJB*Z606D	No
PVLCS Pressure Transmitter	1LSV*V65	1KJB*Z606E	No
Reactor Plant Vent. ΔP Trans.	1HVR*V18	1KJB*Z606F	No
LPCI A to Reactor	1E12*VF044A	1KJB*Z21A	No
LPCI B to Reactor	1E12*VF044B	1KJB*Z21B	No
SW Rtn Vacuum Release	1SWP*SOV552A(e)	1KJB*Z53A	No
SW Rtn Vacuum Release	1SWP*SOV552B(e)	1KJB*Z53B	No
SW Rtn Vacuum Release	1SWP*SOV552C(e)	1KJB*Z53A	No
SW Rtn Vacuum Release	1SWP*SOV552D(e)	1KJB*Z53B	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
b. Manual Isolation Valves			
1. Primary Containment[(a)] (Continued)			
Feedwater Line	1FWS*MOV7A[(e)]	1KJB*Z3A	Yes
Feedwater Line	1FWS*MOV7B[(e)]	1KJB*Z3B	Yes
HPCS Pump Suction from Supp. Pool	1E22*MOVF015[(e)(j)]	1KJB*Z8	No
HPCS to Reactor	1E22*MOVF004[(b)(e)]	1KJB*Z9, 1DRB*Z10	No
HPCS Min. Flow Bypass	1E22*MOVF012[(e)(j)]	1KJB*Z11	No
Supp. Pool Pumpback Rtn.	1DFR*MOV146[(e)(j)]	1KJB*Z11	No
LPCS Suction from Supp. Pool	1E21*MOVF001[(e)(j)]	1KJB*Z12	No
LPCS to Reactor	1E21*MOVF005[(b)(e)]	1KJB*Z13, 1DRB*Z14	No
RCIC Turbine Exh. to Supp. Pool	1E51*MOVF068[(e)]	1KJB*Z17	No
RCIC Min. Flow Bypass	1E51*MOVF019[(e)(j)]	1KJB*Z18A	No
RHR/RCIC Head Spray	1E51*MOVF013[(b)(e)]	1KJB*Z19, 1DRB*Z130	No
LPCI A to Reactor	1E12*MOVF027A[(e)]	1KJB*Z21A	No
LPCI A to Reactor	1E12*MOVF042A[(e)]	1KJB*Z21A	No
LPCI B to Reactor	1E12*MOVF027B[(e)]	1KJB*Z21B	No
LPCI B to Reactor	1E12*MOVF042B[(e)]	1KJB*Z21B	No
LPCI C to Reactor	1E12*MOVF042C[(e)]	1KJB*Z21C	No
RHR A Hx V&R to Supp. Pool	1E12*MOVF073A[(e)(j)]	1KJB*Z23A	No
RHR B Hx V&R to Supp. Pool	1E12*MOVF073B[(e)(j)]	1KJB*Z23B	No
RHR A Min. Flow Bypass	1E12*MOVF064A[(e)(j)]	1KJB*Z24A	No
LPCS Min. Flow Bypass	1E21*MOVF011[(e)(j)]	1KJB*Z24A	No
Post-Acc. Sample Return	1SSR*SOV139[(e)]	1KJB*Z23B	No
RHR B Min. Flow Bypass	1E12*MOVF064B[(e)(j)]	1KJB*Z24B	No
RHR C Min. Flow Bypass	1E12*MOVF064C[(e)(j)]	1KJB*Z24C	No
RHR A Suction-Supp. Pool	1E12*MOVF004A[(e)(j)]	1KJB*Z25A	No
RHR B Suction-Supp. Pool	1E12*MOVF004B[(e)(j)]	1KJB*Z25B	No
RHR C Suction-Supp. Pool	1E12*MOVF105[(e)(j)]	1KJB*Z25C	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

b. Manual Isolation Valves

1. Primary Containment(a) (Continued)

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
CRD Hydraulic Sys. Sup.	1C11*MOVF083(e)	1KJB*Z29	No
Cont. Hydrogen Purge Outlet	1CPP*MOV104(e)	1KJB*Z33	No
Cont. Hydrogen Purge Outlet	1CPP*MOV105(e)	1KJB*Z33	No
SW Supply	1SWP*MOV507A(e)	1KJB*Z52A	No
SW Supply	1SWP*MOV507B(e)	1KJB*Z52B	No
SW Return	1SWP*MOV81A(e)	1KJB*Z53A	No
SW Return	1SWP*MOV81B(e)	1KJB*Z53B	No
SW Return	1SWP*MOV503A(e)	1KJB*Z53A	No
SW Return	1SWP*MOV503B(e)	1KJB*Z53B	No
Air Sup. for Main Steam SRV	1SVV*MOV1B(e)	1KJB*Z102	No
Air Sup. for Main Steam SRV	1SVV*MOV1A(e)	1KJB*Z103	No
Cont. Hydrogen Purge Sup.	1CPP*SOV140(e)	1KJB*Z31	No
Hydrogen Sample Sup.	1CMS*SOV35D(e)	1KJB*Z601E	No
Hydrogen Sample Sup.	1CMS*SOV31B(e)	1KJB*Z601E	No
Hydrogen Sample Rtn.	1CMS*SOV35B(e)	1KJB*Z601F	No
Hydrogen Sample Rtn.	1CMS*SOV31D(e)	1KJB*Z601F	No
Hydrogen Sample Sup.	1CMS*SOV35C(e)	1KJB*Z605E	No
Hydrogen Sample Sup.	1CMS*SOV31A(e)	1KJB*Z605E	No
Hydrogen Sample Rtn.	1CMS*SOV35A(e)	1KJB*Z605F	No
Hydrogen Sample Rtn.	1CMS*SOV31C(e)	1KJB*Z605F	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
b. Manual Isolation Valves			
2. Drywell(k)			
Service Air Supply	1SAS*V489	1DRB*Z45	No
Instrument Air Supply	1IAS*V79	1DRB*Z47	No
Service Water Supply	1HVN*V542	1DRB*Z54	No
Service Water Supply	1SWP*V205	1DRB*Z54	No
Service Water Supply	1SWP*V206	1DRB*Z55	No
Air Sup. for Main Steam SRV	1SVV*V50	1DRB*Z107	No
Air Sup. for Main Steam SRV	1SVV*V53	1DRB*Z112	No
Recirc. Flow Control Hydr.	1RCS*V132	1DRB*Z152	No
Recirc. Flow Control Hydr.	1RCS*V131	1DRB*Z153	No
Recirc. Flow Control Hydr.	1RCS*V162	1DRB*Z154	No
Recirc. Flow Control Hydr.	1RCS*V156	1DRB*Z155	No
Recirc. Flow Control Hydr.	1RCS*V187	1DRB*Z156	No
Recirc. Flow Control Hydr.	1RCS*V186	1DRB*Z157	No
Recirc. Flow Control Hydr.	1RCS*V217	1DRB*Z158	No
Recirc. Flow Control Hydr.	1RCS*V211	1DRB*Z159	No
Cont Atmos. Monitor Probe	1CMS*SOV34A(e)	1DRB*Z500	No
Cont Atmos. Monitor Probe	1CMS*SOV34B(e)	1DRB*Z430	No
Cont Atmos. Monitor Probe	1CMS*SOV34C(e)	1DRB*Z499	No
Cont Atmos. Monitor Probe	1CMS*SOV34D(e)	1DRB*Z428	No
Cont Atmos. Monitor Probe	1CMS*SOV32A(e)	1DRB*Z333	No
Cont Atmos. Monitor Probe	1CMS*SOV32G(e)	1DRB*Z335	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
c. Other Isolation Valves			
1. Primary Containment[(a)]			
Feedwater Line	1B21*AOVF032A[(c)]	1KJB*Z3A	Yes
Feedwater Line	1B21*VF010A[(b)]	1KJB*Z3A	Yes
Feedwater Line	1B21*AOVF032B[(c)]	1KJB*Z3B	Yes
Feedwater Line	1B21*VF010B[(b)]	1KJB*Z3B	Yes
RWCU Disch. to Condenser	1WCS*RV144	1KJB*Z4	Yes
RWCU Backwash Disch.	1WCS*RV154	1KJB*Z5	Yes
HPCS to Reactor	1E22*AOVF005[(b)(c)]	1KJB*Z9, 1DRB*Z10	No
	1DFR*V181	1KJB*Z11	No
	1DFR*V182	1KJB*Z11	No
HPCS Th. Relief to Supp. Pool	1E22*RVF014[(h)]	1KJB*Z11	No
HPCS Th. Relief to Supp. Pool	1E22*RVF035[(h)]	1KJB*Z11	No
HPCS Th. Relief to Supp. Pool	1E22*RVF039[(h)]	1KJB*Z11	No
LPCS to Reactor	1E21*AOVF006[(b)(c)]	1KJB*Z13, 1DRB*Z14	No
RHR/RCIC Head Spray	1E51*AOVF065[(b)(c)]	1KJB*Z19, 1DRB*Z130	No
RHR/RCIC Head Spray	1E51*AOVF066[(b)(c)]	1KJB*Z19, 1DRB*Z130	No
RHR Shutdown Cooling Sup.	1RHS*V240	1KJB*Z20	No
LPCI C to Reactor	1E12*AOVF041C[(b)(c)]	1KJB*Z21C, 1DRB*Z22C	No
RHR A Hx V&R to Supp. Pool	1RHS*RV3A[(h)]	1KJB*Z23A	No
RHR A Hx V&R to Supp. Pool	1E12*RVF055A[(h)]	1KJB*Z23A	No
RHR A Hx V&R to Supp. Pool	1E12*RVF025A[(h)]	1KJB*Z23A	No
RHR A Hx V&R to Supp. Pool	1E12*RVF017A[(h)]	1KJB*Z23A	No
RHR A Hx V&R to Supp. Pool	1E12*RVF005[(h)]	1KJB*Z23A	No
LPCS Th. Relief to Supp. Pool	1E21*RVF018[(h)]	1KJB*Z23A	No
LPCS Th. Relief to Supp. Pool	1E21*RVF031[(h)]	1KJB*Z23A	No
LPCS Th. Relief to Supp. Pool	1E12*RVF036[(h)]	1KJB*Z23A	No
RHR B Hx V&R to Supp. Pool	1RHS*RV3B[(h)]	1KJB*Z23B	No
RHR B Hx V&R to Supp. Pool	1E12*RVF055B[(h)]	1KJB*Z23B	No
RHR B Hx V&R to Supp. Pool	1E12*RVF025C[(h)]	1KJB*Z23B	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
c. Other Isolation Valves			
1. Primary Containment[(a)] (Continued)			
RHR B Hx V&R to Supp. Pool	1E12*RVF025B[(h)]	1KJB*Z23B	No
RHR B Hx V&R to Supp. Pool	1E12*RVF030[(h)]	1KJB*Z23B	No
RHR B Hx V&R to Supp. Pool	1E12*RVF101[(h)]	1KJB*Z23B	No
RHR B Hx V&R to Supp. Pool	1E12*RVF017B[(h)]	1KJB*Z23B	No
Fuel Pool C&C Disch.	1SFC*V101	1KJB*Z26	No
Fuel Pool C&C Suction	1SFC*V350	1KJB*Z27	No
Fuel Pool Purif. Suction	1SFC*V351	1KJB*Z28	No
CRD Hyd. Sys. Sup.	1C11*VF122	1KJB*Z29	No
Equip. Drain Disch.	1DER*V4	1KJB*Z38	No
Floor Drain Disch.	1DFR*V180	1KJB*Z35	No
Fire Protection Hdr.	1FPW*V263	1KJB*Z41	Yes
Service Air Supply	1SAS*V486	1KJB*Z44	Yes
Instr. Air Supply	1IAS*V80	1KJB*Z46	Yes
RPCCW Supply	1CCP*V118	1KJB*Z48	No
RPCCW Return	1CCP*V160	1KJB*Z49	No
Service Water Supply	1SWP*V174	1KJB*Z52A	No
Service Water Supply	1SWP*V175	1KJB*Z52B	No
Air Sup. for Main Steam SRV	1SVV*V9	1KJB*Z102	No
Air Sup. for Main Steam SRV	1SVV*V31	1KJB*Z103	No
Vent. Chilled Water Rtn.	1HVN*V1316	1KJB*Z131	Yes
Vent. Chilled Water Sup.	1HVN*V541	1KJB*Z132	Yes
Condensate Makeup Sup.	1CNS*V86	1KJB*Z134	Yes

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
c. Other Isolation Valves			
2. Drywell(k)			
Main Steam SRV Disch.	1B21*RVF047A	1DRB*Z136	No
Main Steam SRV Disch.	1B21*RVF041A	1DRB*Z137	No
Main Steam SRV Disch.	1B21*RVF051G	1DRB*Z138	No
Main Steam SRV Disch.	1B21*RVF041L	1DRB*Z139	No
Main Steam SRV Disch.	1B21*RVF047C	1DRB*Z140	No
Main Steam SRV Disch.	1B21*RVF041G	1DRB*Z141	No
Main Steam SRV Disch.	1B21*RVF051C	1DRB*Z142	No
Main Steam SRV Disch.	1B21*RVF041C	1DRB*Z143	No
Main Steam SRV Disch.	1B21*RVF047B	1DRB*Z144	No
Main Steam SRV Disch.	1B21*RVF041B	1DRB*Z145	No
Main Steam SRV Disch.	1B21*RVF051B	1DRB*Z146	No
Main Steam SRV Disch.	1B21*RVF041F	1DRB*z147	No
Main Steam SRV Disch.	1B21*RVF047F	1DRB*Z148	No
Main Steam SRV Disch.	1B21*RVF041D	1DRB*Z149	No
Main Steam SRV Disch.	1B21*RVF047D	1DRB*Z150	No
Main Steam SRV Disch.	1B21*RVF051D	1DRB*Z151	No
LPCI A to Reactor	1E12*AOVF041A(c)	1DRB*Z22A	No
LPCI B to Reactor	1E12*AOVF041B(c)	1DRB*Z22B	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
c. Other Isolation Valves			
2. Drywell[k] (Continued)			
Reactor Bldg. Floor Drain Hdr.	1DFR*V4	1DRB*Z37A	No
Reactor Bldg. Floor Drain Hdr.	1DFR*V3	1DRB*Z37A	No
Reactor Bldg. Floor Drain Hdr.	1DFR*V1	1DRB*Z37B	No
Reactor Bldg. Floor Drain Hdr.	1DFR*V2	1DRB*Z37B	No
Reactor Bldg. Equip. Drain Hdr.	1DER*V14	1DRB*Z40A	No
Reactor Bldg. Equip. Drain Hdr.	1DER*V15	1DRB*Z40A	No
Reactor Bldg. Equip. Drain Hdr.	1DER*V16	1DRB*Z40B	No
Reactor Bldg. Equip. Drain Hdr.	1DER*V17	1DRB*Z40B	No
Service Air Supply	1SAS*V487	1DRB*Z45	No
Instr. Air Supply	1IAS*V78	1DRB*Z47	No
RPCCW Supply	1CCP*V119	1DRB*Z50	No
Service Water Supply	1SWP*RV119	1DRB*Z54	No
SLCS Injection	1C41*VEXF004A	1DRB*Z56	No
SLCS Injection	1C41*VEXF004B	1DRB*Z56	No
SLCS Injection	1C41*VF006	1DRB*Z56	No
SLCS Injection	1C41*VF007	1DRB*Z56	No
RPCCW Return	1CCP*V133	1DRB*Z51	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)
c. Other Isolation Valves			
2. Drywell(k) (Continued)			
Air Sup. for Main Steam SRV	1B21*VF036A	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF036F	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF036G	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF036P	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF039C	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF039H	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF039K	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF039S	1DRB*Z107	No
Air Sup. for Main Steam SRV	1B21*VF036J	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF036L	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF036M	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF036N	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF036R	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF039B	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF039D	1DRB*Z112	No
Air Sup. for Main Steam SRV	1B21*VF039E	1DRB*Z112	No
Recirc. Pump Seal Water Sup.	1B33*VF013A	1DRB*Z133	No
Recirc. Pump Seal Water Sup.	1B33*VF017A	1DRB*Z133	No
Recirc. Pump Seal Water Sup.	1B33*VF013B	1DRB*Z135	No
Recirc. Pump Seal Water Sup.	1B33*VF017B	1DRB*Z135	No
Cont. Atmos. Monitor Probe	1CMS*V41	1DRB*Z427	No
Cont. Atmos. Monitor Probe	1CMS*V40	1DRB*Z501	No

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

NOTES

(a) Subject to test pressure of 7.6 psig.

(b) Also isolates the drywell.

(c) Testable check valve.

(d) Isolates on MS-PLCS air line high flow or MS-PLCS air line header to Main Steam Line low differential pressure.

(e) Receives a remote manual isolation signal.

(f) This line is sealed by the penetration valve leakage control system (PVLCS).

(g) This valve sealed by the main steam positive leakage control system (MS-PLCS).

(h) Not subject to Type C leakage tests. Valve(s) will be included in the Type A test.

(j) Valve is hydrostatically leak tested.

(k) Test pressure not applicable to these valves.

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3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY - OPERATING restore SECONDARY CONTAINMENT INTEGRITY - OPERATING 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

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressures within the Shield Building annulus, the Auxiliary Building and the Fuel Building are less than or equal to 3.0, 0.00, and 0.00 inches of vacuum water gauge, respectively.

b. Verifying at least once per 31 days that:

   1. All secondary containment equipment hatch covers are installed.

   2. The door in each access to the secondary containment is closed, except during normal entry and exit.

   3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the Shield Building Annulus and the Auxiliary Building to greater than or equal to 0.5 and 0.25 inches of vacuum water gauge in less than or equal to 18.5 and 13.5 seconds respectively, and maintains it for 1 hour.

2. Verifying that one Fuel Building ventilation subsystem will draw down the Fuel Building to greater than 0.25 inches of vacuum water gauge in less than or equal to 12.5 seconds, and maintains it for 1 hour.

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3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY - FUEL BUILDING

LIMITING CONDITION FOR OPERATION

3.6.5.2 SECONDARY CONTAINMENT INTEGRITY - FUEL BUILDING shall be maintained.

APPLICABILITY: Operational Conditions*

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY - FUEL BUILDING suspend handling of irradiated fuel in the Fuel Building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 SECONDARY CONTAINMENT INTEGRITY - FUEL BUILDING shall be demonstrated, within 24 hours prior to and at least once per 7 days during handling of irradiated fuel in the Fuel Building, by verifying that:

a. The pressure within the Fuel Building is less than or equal to 0.25 inches of vacuum water gauge.

b. All Fuel Building equipment hatch covers are installed.

c. At least one door in each access to the Fuel Building is closed except for routine entry and exit.

d. All Fuel Building penetrations required to be closed during Fuel Handling accident conditions, except the Fuel Building Ventilation System charcoal filtration system penetrations, are closed by valves, blind flanges, or dampers secured in position.

*When irradiated fuel is being handled in the Fuel Building.

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SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.3 The secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.3-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.3-1.

APPLICABILITY: As shown in Table 3.6.5.3-1.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.3-1 inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and, within 8 hours, either:

a. Restore the inoperable damper(s) to OPERABLE status, or

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

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

d. The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION b. or c. above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

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.

Otherwise, in Operational Condition ##, suspend handling of irradiated fuel in the Fuel Building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.3-1 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 verifying the specified isolation time.

##When irradiated fuel is being handled in the Fuel Building.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months, during COLD SHUTDOWN or REFUELING, by verifying that, on a containment isolation test signal, each isolation damper actuates to its isolation position.

c. By verifying the isolation time to be within its limit when tested pursuant to Specification 4.0.5.

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TABLE 3.6.5.3-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

	DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)	DAMPER GROUP#	APPLICABLE OPERATIONAL CONDITION
1.	Shield Building Annulus Ventilation Exhaust Damper (1HVR*AOD161)	15	12	1, 2, 3
2.	Shield Building Annulus Ventilation Exhaust Damper (1HVR*AOD23A)	15	12	1, 2, 3
3.	Shield Building Annulus Ventilation Exhaust Damper (1HVR*AOD23B)	15	12	1, 2, 3
4.	Auxiliary Building Ventilation Exhaust Damper (1HVR*AOD214)	15	11	1, 2, 3
5.	Auxiliary Building Ventilation Exhaust Damper (1HVR*AOD262)	15	11	1, 2, 3
6.	Auxiliary Building Ventilation Exhaust Damper (1HVR*AOD249)	15	11	1, 2, 3
7.	Auxiliary Building Ventilation Exhaust Damper (1HVR*AOD10A)	15	11	1, 2, 3
8.	Auxiliary Building Ventilation Exhaust Damper (1HVR*AOD10B)	15	11	1, 2, 3
9.	Auxiliary Building Ventilation Supply Damper (1HVR*AOD143)	15	11	1, 2, 3
10.	Auxiliary Building Ventilation Supply Damper (1HVR*AOD164)	15	11	1, 2, 3
11.	Fuel Building Ventilation Supply Damper (1HVF*AOD122)	15	13	1, 2, 3, ##
12.	Fuel Building Ventilation Supply Damper (1HVF*AOD101)	15	13	1, 2, 3, ##
13.	Fuel Building Ventilation Exhaust Damper (1HVF*AOD104)	15	13	1, 2, 3, ##
14.	Fuel Building Ventilation Exhaust Damper (1HVF*AOD137)	15	13	1, 2, 3, ##

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TABLE 3.6.5.3-1 (Continued)

	DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)	DAMPER GROUP#	APPLICABLE OPERATIONAL CONDITION
15.	Fuel Building Ventilation Exhaust Damper (1HVF*AOD102)	15	13	1, 2, 3, ##
16.	Fuel Building Ventilation Exhaust Damper (1HVF*AOD112)	15	13	1, 2, 3, ##

#See Table 3.3.2-1.

##When handling irradiated fuel in the Fuel Building.

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STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one standby gas treatment subsystem inoperable, restore the inoperable subsystem 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

4.6.5.4 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 OPERABLE.

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 penetration and bypass leakage testing acceptance criterion of less than 0.05%, using the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and a system flow rate of 12,500 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, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 12,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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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, for a methyl iodide penetration of less than 0.175%.

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and

      b) Annulus ventilation exhaust high radiation signal.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while the filter train is operating at a flow rate of 12,500 cfm ± 10%.

   3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      a) Manual initiation from the control room, and

      b) Simulated automatic initiation signal.

   4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

   5. Verifying that the heaters dissipate $\geq$ 61 kw when tested in accordance with ANSI N510-1980 at the design supply voltage.

e. Verifying, after each complete or partial replacement of a HEPA filter bank, that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 12,500 cfm ± 10%.

f. Verifying, after each complete or partial replacement of a charcoal adsorber bank, that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 12,500 cfm ± 10%.

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SHIELD BUILDING ANNULUS MIXING SYSTEM

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LIMITING CONDITION FOR OPERATION

3.6.5.5 Two independent Shield Building Annulus Mixing subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one Shield Building Annulus Mixing subsystem inoperable, restore the inoperable subsystem 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

4.6.5.5 Each Shield Building Annulus Mixing subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating the subsystem from the control room and verifying that the subsystem operates for at least 15 minutes and

b. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and

      b) Annulus ventilation exhaust high radiation signal.

   2. Verifying that each subsystem has a flow rate of 52,500 cfm ± 10%.

   3. Verifying that the subsystem starts and isolation dampers open on each of the following test signals:

      a) Manual initiation from the control room, and

      b) Simulated automatic initiation signal.

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FUEL BUILDING VENTILATION

LIMITING CONDITION FOR OPERATION

3.6.5.6 Two independent Fuel Building Ventilation Charcoal Filtration subsystems shall be OPERABLE and one operating in the emergency mode when in Operational Condition *.

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

ACTION:

a. With one Fuel Building Ventilation Charcoal Filtration 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 *, suspend handling of irradiated fuel in the Fuel Building. The provisions of Specification 3.0.3 are not applicable.

b. With both Fuel Building Ventilation Charcoal Filtration subsystems inoperable or with one not operating in the emergency mode in Operational Condition *, suspend handling of irradiated fuel in the Fuel Building. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.6 Each Fuel Building Ventilation Charcoal Filtration subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours, in Operational Condition *, by verifying one Fuel Building Ventilation Charcoal Filtration subsystem in operation.

b. 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 OPERABLE.

*When irradiated fuel is being handled in the Fuel Building.

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 the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05%, using the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and a system flow rate of 10,000 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, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 10,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation, by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

e. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and

      b) Fuel Building ventilation exhaust high radiation signal.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches water gauge while the filter train is operating at a flow rate of 10,000 cfm ± 10%.

   3. Verifying that the subsystem starts and isolation dampers actuate to isolate the normal flow path and to divert flow through the charcoal filters on each of the following test signals:

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SURVEILLANCE REQUIREMENTS (Continued)

a) Manual initiation from the control room, and

b) Simulated automatic initiation signal.

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5. Verifying that the heaters dissipate $\geq$49 kw when tested in accordance with ANSI N510-1980 at the design supply voltage.

f. Verifying, after each complete or partial replacement of a HEPA filter bank, that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 10,000 cfm ± 10%.

g. Verifying, after each complete or partial replacement of a charcoal adsorber bank, that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criterion of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 10,000 cfm ± 10%.

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CONTAINMENT SYSTEMS

3/4.6.6 ATMOSPHERE CONTROL

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PRIMARY CONTAINMENT HYDROGEN RECOMBINER SYSTEMS

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 primary containment 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 primary containment 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 control room recombiner indication instrumentation and control circuits.

   2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e., loose wiring or structural connections, deposits of foreign materials, etc.

   3. Verifying the integrity of all heater electrical circuits by performing a resistance-to-ground test within 30 minutes 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, during a recombiner system functional test, that the heater sheath temperature increases to greater than or equal to 1215°F within 5 hours and is maintained for at least 4 hours.

PRIMARY CONTAINMENT/DRYWELL HYDROGEN MIXING SYSTEM

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LIMITING CONDITION FOR OPERATION

3.6.6.2 Two primary containment/drywell hydrogen mixing systems shall be OPERABLE and the inlet and outlet valves shall be closed, except one inlet or outlet 6-inch valve may be opened for controlling drywell pressure with the following time limits:

a. In OPERATIONAL CONDITION 3, not to exceed 90 hours per 365 days, and

b. In OPERATIONAL CONDITION 1 or 2, not to exceed 5 hours per 365 days.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one hydrogen mixing system inlet or outlet valve open during OPERATIONAL CONDITIONS 1 and 2 for more than 5 hours per 365 days, immediately close the hydrogen mixing valves or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one hydrogen mixing system inlet or outlet valve open during OPERATIONAL CONDITION 3 for more than 90 hours per 365 days, immediately close the hydrogen mixing valves or be in at least COLD SHUTDOWN within the next 24 hours.

c. With one primary containment/drywell 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.

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SURVEILLANCE REQUIREMENTS

4.6.6.2 Each primary containment/drywell hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 7 days, by determining the cumulative time that:

   1. The hydrogen mixing system inlet or outlet valve has been open during OPERATIONAL CONDITIONS 1 and 2 during the past 365 days, and

   2. The hydrogen mixing system inlet or outlet valve has been open during OPERATIONAL CONDITION 3 during the past 365 days.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by:

   1. Starting the system from the control room, and

   2. Verifying that the system operates for at least 15 minutes.

c. At least once per 18 months by verifying a system flow rate of at least 600 cfm.

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PRIMARY CONTAINMENT/DRYWELL HYDROGEN IGNITION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 The primary containment/drywell hydrogen ignition system, consisting of two independent primary containment/drywell hydrogen ignition subsystems each consisting of ten circuits, shall be OPERABLE with no more than two igniter assemblies inoperable per circuit and no more than five igniter assemblies inoperable per subsystem and no adjacent igniter assemblies inoperable.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION:

a. With one primary containment/drywell hydrogen ignition subsystem and/or circuit inoperable, restore the inoperable subsystem and/or circuit to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With any adjacent igniter assemblies inoperable, restore all igniter assemblies adjacent to an inoperable igniter assembly to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The primary containment/drywell hydrogen ignition system shall be demonstrated OPERABLE:

a. At least once per 184 days by energizing all the igniter assemblies and performing current/voltage measurements of each circuit.

   1. If more than three igniter assemblies on either subsystem are determined to inoperable, Specification 4.6.6.3.a shall be performed at least once per 92 days until this condition no longer exists.

   2. If more than one igniter assembly on each subsystem are determined to be inoperable, determine if the inoperable igniter assemblies are adjacent.

b. At least once per 18 months, by energizing each igniter assembly, verifying a surface temperature of at least 1700°F for each of the accessible igniters and verifying by measurement sufficient current/voltage to develop 1700°F surface temperature for those igniter assemblies in inaccessible areas.

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3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least two independent standby service water (SSW) system subsystems, with each subsystem comprised of:

a. Two OPERABLE SSW pumps, and

b. An OPERABLE flow path capable of taking suction from the standby cooling tower basin and transferring the water through associated systems and components required to be OPERABLE,

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5 and*, the subsystem(s) associated with systems and components required OPERABLE by Specifications 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1, and 3.9.11.2.

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

ACTION:

a. With the SSW flow path to one or more systems or components inoperable, declare the associated system or component inoperable and take the required action.

b. In OPERATIONAL CONDITION 1, 2, or 3:

   1. With one SSW 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.

   2. With one SSW pump in each subsystem inoperable restore at least one 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 SSW subsystem otherwise inoperable, restore the inoperable subsystem 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.

*When handling irradiated fuel in the primary containment or Fuel Building.

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ACTION: (Continued)

4. With both SSW subsystems otherwise inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN** within the following 24 hours.

c. With only one SSW pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or:

1. In OPERATIONAL CONDITION 4 or 5, declare the associated equipment inoperable and take the ACTION required by Specifications 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1, and 3.9.11.2.

2. In Operational Condition *, verify adequate cooling for the diesel generators required to be OPERABLE or declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system subsystem(s) 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.

b. At least once per 18 months during shutdown by verifying that each automatic valve actuates to the correct position and each pump starts on a normal service water low-pressure signal.

*When handling irradiated fuel in the primary containment or Fuel Building.
**Whenever both RHR shutdown cooling mode loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.2 The standby cooling water storage basin shall be OPERABLE with:

a. A minimum basin water level at or above elevation 111'10" Mean Sea Level, USGS datum, and

b. A basin water temperature of less than or equal to 82°F.

c. Two OPERABLE cooling tower fan cells (5 fans per cell) per division.

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

ACTION:

With the requirements of the above specification not satisfied:

a. With the basin water level less than 111'10" MSL or the temperature greater than 82°F, then declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1.

b. In OPERATIONAL CONDITION 1, 2, or 3 with any one fan cell inoperable, restore the inoperable fan cell to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the next 24 hours.

c. In OPERATIONAL CONDITION 1, 2, or 3 with one fan cell per division inoperable, restore at least one to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

d. In OPERATIONAL CONDITION 1, 2, or 3 with both fan cells in one division inoperable, restore at least one of the inoperable fan cells to OPERABLE status within 72 hours and with both SSW pumps in the other division inoperable, align the OPERABLE SSW pumps to the OPERABLE fan cells within 2 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. In OPERATIONAL CONDITION 1, 2, or 3 with the cooling tower fan cells otherwise inoperable be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

f. In OPERATIONAL CONDITION 4, 5, * with one or less fan cells OPERABLE, declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

*When handling irradiated fuel in the primary containment or Fuel Building.

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SURVEILLANCE REQUIREMENTS

4.7.1.2 The standby cooling tower and water storage basin shall be determined OPERABLE:

a. At least once per 24 hours by verifying the basin water level to be at least elevation 111'10".

b. By verifying the average water temperature to be less than or equal to 82°F:

   1. At least once per 24 hours, or

   2. At least once per 4 hours when the last recorded basin water temperature is greater than or equal to 75°F, or

   3. At least once per 2 hours when the last recorded basin water temperature is greater than or equal to 80°F.

c. At least once per 31 days by starting the cooling tower fans in each cell from the control room and operating each fan cell for at least 15 minutes.

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3/4.7.2 MAIN CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The main control room air conditioning system, with two independent air handling unit/filter train subsystems, shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one main control room air conditioning subsystem inoperable, restore the inoperable subsystem 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.

b. In OPERATIONAL CONDITION 4, 5 or *:

   1. With one main control room air conditioning air handling/filter train subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the emergency mode of operation.

   2. With both main control room air conditioning air handling/filter train subsystems inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and Fuel Building and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

SURVEILLANCE REQUIREMENTS

4.7.2 Each main control room air conditioning 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 104°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

*When irradiated fuel is being handled in the primary containment or Fuel Building.

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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 the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criterion of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

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, for a methyl iodide penetration of less than 0.175%.

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 8 inches water gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%.

   2. Verifying that on each of the below emergency mode actuation test signals, the subsystem automatically switches to the emergency mode of operation, the isolation valves close within 30 seconds, and the control room is maintained at a positive pressure of $\geq$ 1/8 inch water gauge relative to the outside atmosphere during subsystem operation at a flow rate less than or equal to 4,000 cfm:

      a) LOCA, and

      b) Local air intake radiation monitor - High.

   3. Verifying that the heaters dissipate 23 ± 2.3 kw when tested in accordance with ANSI N510-1980, at the design supply voltage.

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SURVEILLANCE REQUIREMENTS (Continued)

f. Verifying after each complete or partial replacement of a HEPA filter bank that the HEPA filter bank removes 99.95% of the DOP when they are tested in place in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ± 10%.

g. Verifying after each complete or partial replacement of a charcoal adsorber bank that the charcoal adsorber bank removes 99.95% of a halogenated hydrocarbon refrigerant test gas when tested in place in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ± 10%.

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3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 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 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue 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 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

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.

3. Verifying that the pump flow controller is in the correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to steam being supplied at a reactor pressure of 1020 + 25, - 100 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 test.

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and restart 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.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 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 water level-low signal and on a suppression pool water level - high 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.

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3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement 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.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3, and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are OPERABLE, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are OPERABLE. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.4.f.

d. Transient Event Inspection

An inspection shall be performed of all mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients, as determined from a review of operational data or a visual inspection of the systems, within 72 hours for accessible areas and within 6 months for inaccessible areas, following this determination. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

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

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing, prior to the test period, of the sample plan selected, or the sample plan used in the prior test period shall be implemented:

1. At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.4.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2. A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3. An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type been tested.

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4. For each type of snubber, an initial representative sample of 88 snubbers shall be functionally tested. Three snubbers of each type are allowed not meeting the functional test acceptance criteria. If the number of snubbers that failed the test is greater than 3, an additional sample of that type of snubber equal to 22(A-3) shall be functionally tested, where "A" is the total number of snubbers failed during the functional test of the representative sample. For each snubber that failed during the functional test of the resample, an additional sample of 22 snubbers of the same type shall be functionally tested. The tests shall continue until no more failures are found or until all snubbers of that type have been functionally tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that the sample is representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If, during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression;

2. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

3. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

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

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g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers, irrespective of type, which may be subject to the same failure mode.

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

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.4.e for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers that fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers that have repairs that might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months from the time the snubber was installed in the unit.

i. Snubber Service Life Program

The service life of all snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

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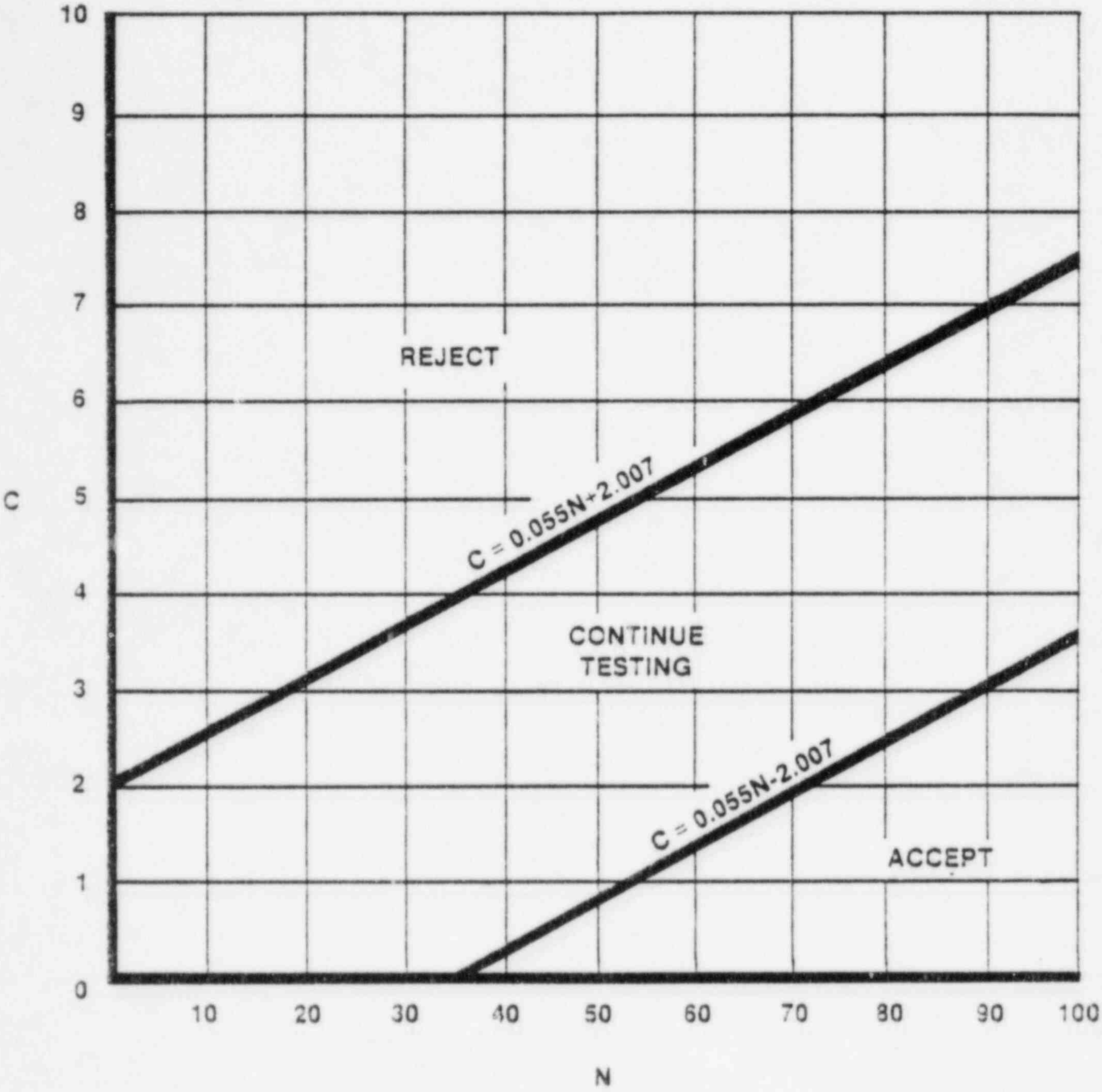

FIGURE 4.7.4-1
SAMPLE PLAN FOR SNUBBER FUNCTIONAL TEST

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3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material in excess of either 100 microcuries of beta and/or gamma emitting material or 10 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

   1. Decontaminate and repair the sealed source, or

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

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 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.

4.7.5.2 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4.7.5.3 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.

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3/4.7.6 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The fire suppression water system shall be OPERABLE with:

a. Three fire suppression pumps, each with a capacity of 1500 gpm, with their discharges aligned to the fire suppression header,

b. Two separate fire water tanks, each with a minimum contained volume of 253,000 gallons, and

c. An OPERABLE flow path capable of taking suction from both water storage tanks 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 3.7.6.5, 3.7.6.4, and 3.7.6.2.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or 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, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 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 by starting the 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.

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d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. 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 fire suppression pump develops at least 2250 gpm at a system head of 248 feet,

2. Cycling each valve in the flow path, that is not testable during plant operation, through at least one complete cycle of full travel, and

3. Verifying that each fire suppression pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 70 psig.

f. 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.

4.7.6.1.2 Each diesel driven fire suppression pump shall be demonstrated OPERABLE:

a. At least once per 31 days by;

1. Verifying the fuel day tank contains at least 300 gallons of fuel.

2. Starting the pump from ambient conditions and operating it for greater than or equal to 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM D270-75, is within the acceptable limits specified in Table 1 of ASTM D975-77 when checked for viscosity, water, and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.6.1.3 Each diesel driven fire pump 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 cell is above the plates, and

   2. The overall battery bank voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery. The specific gravity, corrected to 77°F and full electrolyte level, shall be greater than or equal to 1.200.

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

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

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

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3.7.6.2 The following spray and sprinkler systems shall be OPERABLE:

LOCATION	ELEVATION	SYSTEM IDENTITY
a. Control Bldg. Cable Chases	116'0"	AS-6A
	98'0"	AS-6B
	70'0"	AS-6C, WS-6A, WS-6B, WS-6C
	115'0"	WS-7A, WS-7B
b. Cable Tunnels	67'6"/70'0"	WS-8D, WS-8E, WS-8F,
	67'6"/70'0"	WS-8G, WS-8H, WS-8K,
	67'6"/70'0"	WS-8L, WS-8M, WS-8N
c. Auxiliary Bldg., RCIC	70'0"	PS-1, WS-19
Pump Room	141'0"	WS-4A, WS-4B, WS-20, AS-12
d. Diesel Generator Bldg.	98'0"	PS-2A, PS-2B, PS-2C
e. Fuel Bldg.	95'0"	AS-5
	148'0"	WS-5A, WS-5B

APPLICABILITY: Whenever equipment protected by the spray or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray or sprinkler 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.

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

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4.7.6.2 Each of the above required spray and 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 automatic systems, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation test signal, and

      b) Cycling each valve in the flow path, that is not testable during plant operation, through at least one complete cycle of full travel.

   2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity.

   3. By a visual inspection of each deluge nozzle's spray area* to verify that the spray pattern is not obstructed.

d. At least once per 3 years by performing an air or water flow test through each open head spray and sprinkler header system* and verifying that each open head spray nozzle and sprinkler header system is unobstructed.

*The charcoal filter system spray nozzles need only be visually inspected and verified to be unobstructed each time the charcoal is changed.

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LIMITING CONDITION FOR OPERATION

3.7.6.3 The following main control room Power Generation Control Complex (PGCC) Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure:

PGCC Panel Module U701	PGCC Panel Module U730
PGCC Panel Module U702	PGCC Panel Module U731
PGCC Panel Module U703	PGCC Panel Module U732
PGCC Panel Module U704	PGCC Panel Module U740
PGCC Panel Module U710	PGCC Panel Module U741
PGCC Panel Module U711	PGCC Panel Module U742
PGCC Panel Module U712	PGCC Panel Module U743
PGCC Panel Module U713	PGCC Panel Module U744
PGCC Panel Module U714	PGCC Panel Module U745
PGCC Panel Module U715	PGCC Panel Module U746
PGCC Panel Module U717	PGCC Panel Module U747
PGCC Panel Module U720	PGCC Panel Module U748
PGCC Panel Module U721	PGCC Panel Module U799
PGCC Panel Module U723	PGCC Panel Module U750

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 1 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.

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

SURVEILLANCE REQUIREMENTS

4.7.6.3 The above required Halon system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each Halon system storage tank is pressurized to at least 280 psig.

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

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Verifying the system actuates, manually and automatically, upon receipt of a simulated actuation signal (actual Halon release, Halon bottle initiator valve actuation, and electro-thermal link burning may be excluded from the test), and

2. Performance of a flow test through headers and nozzles to assure no blockage.

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.6.4 The fire hose stations shown in Table 3.7.6.4-1 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 3.7.6.4-1 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.6.4 Each of the fire hose stations shown in Table 3.7.6.4-1 shall be demonstrated OPERABLE:

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

b. At least once per 18 months by:

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

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years by:

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

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

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TABLE 3.7.6.4-1

FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK INDENTIFICATION
a.	Reactor Building	114'0"	HR - 16, 22
		141'0"	HR - 17, 23
		162'3"	HR - 18, 19, 24, 25
		186'3"	HR - 20, 21, 26
b.	Auxiliary Building	70'0" (Stairwell)	HR - 84
		95'9"	HR - 6, 7, 8, 9
		114'0"	HR - 10, 11
		141'0"	HR - 12, 13, 14, 15
		170'0"	HR - 80
c.	Control Building	70'0"	HR - 85, 86, 87
		98'0"	HR - 88, 89, 90
		115'0" and 116'0"	HR - 91, 92, 93, 94
		135'0" (Stairwell)	HR - 96
d.	Fuel Building	70'0"	HR - 1, 2, 82
		95'0"	HR - 3, 4
		113'0"	HR - 81
		148'0"	HR - 5
e.	Pipe Tunnel	67'6"	HR - 83
f.	Turbine Building	95'0"	HR - 50, 51
		123'6"	HR - 53

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YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.6.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7.6.5-1 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 Table 3.7.6.5-1 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise provide the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.6.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7.6.5-1 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, 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 of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

   2. Replacement of all degraded gaskets in couplings.

   3. Performing a flow check of each hydrant.

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TABLE 3.7.6.5-1

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION		HYDRANT NUMBER
a.	Northeast of Fuel Bldg	FHY 11
b.	East of Control Bldg	FHY 13
c.	West of Standby Cooling Tower	FHY 9*
d.	North of Fuel Bldg	FHY 10

*No associated hose house.

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3/4.7.7 FIRE-RATED ASSEMBLIES

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LIMITING CONDITION FOR OPERATION

3.7.7 All fire barrier assemblies shall be OPERABLE. Fire barrier assemblies include:

a. Walls, floors/ceilings, cable tray enclosures, and other fire barriers that separate safety-related fire areas or that separate portions of redundant systems, important to safe shutdown, within a fire area, and

b. All sealing devices in fire-rated assembly penetrations, including fire doors and fire dampers and cable, piping and ventilation duct penetration seals, and ventilation seals.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire-rated assemblies or sealing devices inoperable, within 1 hour establish a continuous fire watch on at least one side of the affected assembly and/or sealing device or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly or sealing device, and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each of the above required fire-rated assemblies and penetration sealing devices shall be verified OPERABLE at least once per 18 months by performing a visual inspection of:

a. The exposed surfaces of each fire-rated assembly.

b. Each fire damper and associated hardware.

c. At least 10 percent of each type of sealed penetration. If changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample is found with no apparent changes in appearance or abnormal degradation. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. At least once per 31 days, the OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST.

b. At least once per 7 days, that each locked-closed fire door is closed.

c. At least once per 24 hours, that doors with automatic hold-open and release mechanisms are free of obstructions and, at least once per 18 months, by performing a functional test of these mechanisms.

d. At least once per 24 hours, that each unlocked fire door without electrical supervision is closed.

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3/4.7.8 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The temperature of each area shown in Table 3.7.8-1 shall be maintained within the limits indicated in Table 3.7.8-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.8-1:

a. For more than 8 hours, prepare and submit, within the next 30 days, a Special Report to the Commission, pursuant to Specification 6.9.2, providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.8 At least once per 12 hours, the temperature in each of the areas shown in Table 3.7.8-1 shall be determined to be within its limit.

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TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1. Auxiliary Building	
a. LPCS area	122
b. RHR A pump room	122
c. RCIC pump room	122
d. RHR B pump room	122
e. RHR C pump room	122
f. HPCS pump room	122
g. MCC area (West)	112
h. MCC area (East)	116
i. Main steam tunnel (north)	122
j. Standby gas treatment rooms	122
k. Annulus mixing fan area	122
l. RHR Hx Area (West)	122
m. Hoist Area	122
n. RHR Hx Area (East)	122
o. HPCS Hatch Area	122
p. RPCCW Area	122
q. Elevator Room	122
r. RPCCW Area	122
s RHR Equip. Removal Cubicles	122
2. Diesel Generator Control Rooms	
a. Diesel Generator 1A	104
b. Diesel Generator 1B	104
c. Diesel Generator 1C	104
3. Control Building	
a. Standby switchgear room 1A	104
b. Standby switchgear room 1B	104
c. Division I battery room	90
d. Division II battery room	90
e. Division III battery room	90
f. Inverter 1A room	104
g. Inverter 1B room	104
h. Inverter 1C room	104

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3/4.7.9 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.9 The main turbine bypass system shall be OPERABLE.

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

ACTION:

With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or be at less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.9 The main turbine bypass system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

2. Demonstrating that TURBINE BYPASS SYSTEM RESPONSE TIMES meet the following requirements when measured from the initial closure movement of the main turbine stop valve or the main turbine control valve:

   a) Main turbine bypass valve opening shall start within 0.1 seconds, and

   b) at least 80% of main turbine bypass capacity shall be established within 0.3 seconds.

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3/4.7.10 STRUCTURAL SETTLEMENT

LIMITING CONDITION FOR OPERATION

3.7.10 Structural settlement shall be within the predicted values as shown in Table 3.7.10-1 and calculated differential settlements shall be within the allowable ranges shown in Table 3.7.10-2 for the following structures:

a. Reactor Building
b. Auxiliary Building
c. Fuel Building
d. Control Building
e. Diesel Generator Building
f. Standby Cooling Tower, Basin and Pump House
g. BF Tunnel
h. Main Steam Tunnel
i. E Tunnel
j. G Tunnel

APPLICABILITY: At all times.

ACTION:

With the measured structual settlement of any of the above required structures outside of the limits of Tables 3.7.10-1 and 3.7.10-2, prepare and submit, within the next 30 days, a Special Report to the Commission, pursuant to Specification 6.9.2, providing a record of the settlement measurements and the predicted settlement, an analysis to demonstrate the continued structural integrity of the affected structure(s), and plans to monitor the settlement of the affected structure(s) in the future.

SURVEILLANCE REQUIREMENTS

4.7.10 The structural settlement of the above required structures shall be demonstrated to be within the limits of Tables 3.7.10-1 and 3.7.10-2:

a. At least once per 92 days, until there is essentially no movement during those 92 days.

b. At least once per 24 months, for at least 10 years.

c. Following any seismic event equal to or greater than an Operational Basis Earthquake (OBE).

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TABLE 3.7.10-1

TOTAL PREDICTED SETTLEMENTS OF MAJOR STRUCTURES

STRUCTURE	SETTLEMENT MARKER NO.	PREDICTED SETTLEMENT (IN.)
Reactor Building	15	4.0
	16	4.0
	17	4.0
Auxiliary Building	18	3.8
	19	3.6
	20	3.9
	21	3.7
Fuel Building	11	3.7
	12	4.0
	13	3.5
	14	3.8
Control Building	5	3.7
	6	3.3
	7	3.7
	8	3.7
Diesel Generator Building	1	3.4
	2	3.7
	3	3.6
	4	3.8
Standby Cooling Tower, Basin and Pump House	30	2.7
	31	3.2
	32	2.4
BF Tunnel	9	2.1
	10	2.5
Main Steam Tunnel	22	3.8
	23	3.8
E Tunnel	28	3.8
	29	2.8
G Tunnel	33	2.6
	34	0.4

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TABLE 3.7.10-2

ALLOWABLE DIFFERENTIAL SETTLEMENTS OF MAJOR STRUCTURAL INTERFACE POINTS

Building Interface	Marker No. A	Marker No. B	Allowable Differential Settlement (in.)
Diesel Generator vs. Control	2	5	+0.35 to -0.39
	4	7	+0.42 to -0.61
BF Tunnel vs. Diesel Generator	9	3,4*	+0.53 to -1.08
BF Tunnel vs. Fuel	10	12	+0.56 to -1.34
Fuel vs. Reactor	12	15	+0.26 to -0.61
	14	17	+0.30 to -0.60
Reactor vs. Auxiliary	16	18	+0.32 to -0.08
	17	20	+0.33 to -0.13
Auxiliary vs. Main Steam	19,21*	22	+0.44 to -0.69
Fuel vs. G Tunnel	13	33	+0.41 to -0.32
Fuel vs. E Tunnel	14	28	+0.42 to -0.39
E Tunnel vs. Auxiliary	29	21	+0.73 to -0.43
Control vs. Auxiliary	7	18	+0.46 to -0.66
	8	19	+0.50 to -0.50

NOTE: Positive differential settlement indicates settlement of Marker A with respect to Marker B. Negative sign indicates settlement of Marker B with respect to Marker A.

*Settlements for these two markers should be averaged when determining differential settlement.

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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. A separate day fuel tank containing a minimum of 316.3 gallons of fuel,

   2. A separate fuel storage system containing a minimum of 45,495 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 Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If a diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours* for each diesel generator which has not been successfully tested within the past 24 hours. 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 Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If a diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours* for each diesel generator which has not been successfully tested within the past 24 hours. Restore at least one of the inoperable A.C.

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

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ACTION (Continued)

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.

c. 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 Requirement 4.8.1.1.1.a within 1 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 and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.1 and 3.7.1.1.

d. With diesel generator 1A or 1B of the above required A.C. electrical power sources inoperable, in addition to ACTION a or b, as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and the C standby service water pump OPERABLE if diesel generator 1B is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. 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 8 hours 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.

f. 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 Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter and Surveillance Requirement 4.8.1.1.2.a.4 for diesel generator 1C within 8 hours.* 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.

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

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. At least once per 7 days, determined OPERABLE by verifying correct breaker alignments and indicated power availability, and

b. At least once per 18 months, demonstrated OPERABLE during shutdown by manually transferring unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators 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 fuel tank.

2. Verifying the fuel level in the fuel storage tank.

3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 450 rpm for diesel generators 1A and 1B and 900 rpm for diesel generator 1C in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual.
b) Simulated loss of offsite power by itself.
c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
d) An ESF actuation test signal by itself.

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*All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures (e.g. gradual acceleration and/or gradual loading > 150 sec.) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

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5. Verifying the diesel generator is synchronized, loaded to 3030-3130 kw** for diesel generators 1A and 1B and 2500-2600 kw** for diesel generator 1C in less than or equal to 60 seconds, and operates with this load for at least 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 160 psig.

b. At least once per 24 hours by verifying for diesel generator 1A and 1B that the lube oil circulating pump is operating.

c. By removing accumulated water:

1. From the day tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and

2. From the storage tank at least once per 31 days.

d. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1. By verifying in accordance with the tests specified in ASTM D975-81, prior to addition to the storage tanks, that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or qual to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

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2. By verifying an antioxidant type diesel fuel oil stabilizer is added to new fuel added to the storage tanks in accordance with manufacturer's recommendations.

3. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81, except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78 and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

f. 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 917.5 kw for diesel generator 1A, greater than or equal to 509.2 kw for diesel generator 1B, and greater than or equal to 1995 kw for diesel generator 1C while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz, and while maintaining engine speed less than 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 3030-3130 kw** for diesel generators 1A and 1B and 2500-2600 kw** for diesel generator 1C without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

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

a) For divisions I and II:

1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the sequencing logic and operates for greater than or equal to 5 minutes while its generator is loaded with the loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

b) For division III:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads within 10 seconds, energizes the auto-connected loads through the sequence logic, and operates for greater than or equal to 5 minutes while its generator is loaded with the loads. After energization, the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 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 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

a) For divisions I and II:

1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the sequencing logic and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

b) For division III:

1) Verifying de-energization of the emergency bus.

2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its permanently connected loads within 10 seconds, energizes the auto-connected loads through the sequencing logic and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

7. Verifying that all automatic diesel generator trips are automatically bypassed upon an ECCS actuation signal except engine overspeed and generator differential current.

8. Verifying the diesel generator operates for at least 24 hours. During the first 22 hours of this test, the diesel generator shall be loaded to 3030-3130** kw for diesel generators 1A and 1B and 2500-2600** kw for diesel generator 1C. During the remaining 2 hours of this test, the diesel generator shall be loaded to 3030-3130** kw for diesel generator 1A and 1B and 2750-2850** kw for diesel generator 1C. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.f.4.a)2) and b)2)*.

9. Verifying that the auto-connected loads to each diesel generator do not exceed 3130 kw for diesel generator 1A and 1B and 2600 kw for diesel generator 1C.

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.

*If Surveillance Requirements 4.8.1.1.2.f.4.a)2) and b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, perform Surveillance Requirement 4.8.1.1.2.a.5 prior to repeating Surveillance Requirements 4.8.1.1.2.f.4.a)2) and b)2) for the appropriate diesel.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

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

12. Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 1A, 1B and 1C.

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

    a) For Diesel Generators 1A and 1B:
        1) Loss of control power to diesel control panel.
        2) Starting air pressure below 150 psi.
        3) Stop solenoid energized.
        4) Diesel in the maintenance mode (includes barring device engaged).
        5) Overspeed trip device actuated.
        6) Generator backup protection lockout relay tripped.

    b) For Diesel Generator 1C:
        1) Diesel generator lockout relays not reset.
        2) Diesel engine mode switch not in "AUTO" position.
        3) Diesel generator output breaker closed before start of diesel.
        4) Diesel generator output breaker in racked-out position.
        5) *Diesel generator regulator mode switch not in "AUTO" position.
        6) Insufficient starting air pressure.
        7) Loss of dc power to diesel generator controls.

g. Verify the Division III diesel generator ambient room temperature to be $\geq$ 45°F:

    1. At least once per 24 hours with the last reported room temperature $\geq$ 50°F, or

    2. At least once per 12 hours with the last reported room temperature < 50°F.

h. At least once per 10 years, or after any modifications which could affect diesel generator interdependence, by starting all three diesel generators simultaneously, during shutdown, and verifying that all three diesel generators accelerate to at least 450 rpm for diesel generators 1A and 1B and 900 rpm for diesel generator 1C in less than or equal to 10 seconds.

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*Item 5) does not electrically block diesel generator from emergency starting; however, it will affect the loading and operation of the diesel.

i. At least once per 10 years by:

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

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code, in accordance with ASME Code Section 11 Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission, pursuant to Specification 6.9.2, within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

TABLE 4.8.1.1.2-1

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DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Number of Failures in Last 100 Valid Tests*	Test Frequency
$\leq$ 1	$\leq$ 4	Once per 31 days
$\geq$ 2**	$\geq$ 5	Once per 7 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, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with Surveillance Requirement 4.8.1.1.2.a.5, four tests, in accordance with the 184-day testing of Surveillance Requirement 4.8.1.1.2.a.4. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

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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. A day fuel tank containing a minimum of 316.3 gallons of fuel,

   2. A fuel storage system containing a minimum of 45,495 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment, operations with a potential for draining the reactor vessel, and crane operations over the spent fuel storage pool when fuel assemblies are therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 23 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator 1C of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.2, 3.5.3 and 3.7.1.1.

c. The provisions of Specification 3.0.3 are not applicable.

*When handling irradiated fuel in the primary containment or Fuel Building.

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4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 (except for the requirement of 4.8.1.1.2.a.5), and 4.8.1.1.3 .

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D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division I, consisting of:
   1. 125 volt battery 1A.
   2. 125 volt full capacity Class 1E source charger.

b. Division II, consisting of:
   1. 125 volt battery 1B.
   2. 125 volt full capacity Class 1E source charger.

c. Division III, consisting of:
   1. 125 volt battery 1C.
   2. 125 volt full capacity Class 1E source charger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division I or Division II battery and/or charger of the above required D.C. electrical power sources 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 battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system and the associated SSW subsystem inoperable and take the ACTION required by Specifications 3.5.1 and 3.7.1.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

   2. Total battery terminal voltage is greater than or equal to 130.2 volts on float charge.

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b. At least once per 92 days, and within 7 days after a battery discharge with battery terminal voltage below 110 volts or after a battery overcharge with battery terminal voltage above 144 volts, by verifying that:

1. The parameters in Table 4.8.2.1-1 meet the Category B limits,

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

3. The average electrolyte temperature of at least one out of six connected cells is above 60°F.

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

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

2. The cell-to-cell and terminal connections are clean, tight, 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 150×10^{-6} ohms and

4. The battery charger will supply at least 300 amperes for chargers 1A and 1B and 50 amperes for charger 1C at a minimum of 130.2 volts for at least 8 hours.

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 the design duty cycle 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 in accordance with IEEE 450 while maintaining the battery terminal voltage greater than or equal to 105 volts.

a) Division I

$\geq$ 671 amperes for the first 60 seconds
$\geq$ 270 amperes for the next 9 minutes
$\geq$ 336 amperes for the next 60 seconds
$\geq$ 270 amperes for the next 228 minutes
$\geq$ 451 amperes for the last 60 seconds

SURVEILLANCE REQUIREMENTS (Continued)

b) Division II
$\geq$ 502 amperes for the first 60 seconds
$\geq$ 261 amperes for the next 9 minutes
$\geq$ 327 amperes for the next 60 seconds
$\geq$ 261 amperes for the next 228 minutes
$\geq$ 327 amperes for the last 60 seconds

c) Division III
$\geq$ 53.2 amperes for the first 60 seconds
$\geq$ 15.4 amperes for the next 120 minutes

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

f. At least once per 18 months, during shutdown, performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark and $\leq$ ¼" above maximum level indication mark	Above top of plates and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts[c]	$\geq$ 2.07 volts
Specific Gravity[a]	$\geq$ 1.200[b] (Div. I&II) $\geq$ 1.195[b] (Div. III)	$\geq$ 1.195 (Div. I&II) $\geq$ 1.190 (Div. III)	Not more than .020 below the average of all connected cells
		Average of all connected cells $\geq$ 1.205 (Div. I&II) $\geq$ 1.200 (Div. III)	Average of all connected cells $\geq$ 1.195[b] (Div. I&II) $\geq$ 1.190[b] (Div. III)

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amperes when on float charge.

(c) May be corrected for average electrolyte temperature.

(1) For Category A parameters outside the limits shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameters are restored to within limits within the next 6 days.

(2) For Category B parameters outside the limits shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameters are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

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D.C. SOURCES - 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 D.C. electrical power sources shall be OPERABLE with:

a. Division I consisting of:
   1. 125 volt battery 1A.
   2. 125 volt full capacity Class 1E source charger.

b. Division II consisting of:
   1. 125 volt battery 1B.
   2. 125 volt full capacity Class 1E source charger.

c. Division III consisting of:
   1. 125 volt battery 1C.
   2. 125 volt full capacity Class 1E source charger.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the division I and/or division II battery and/or charger of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment or Fuel Building, and operations with a potential for draining the reactor vessel.

b. With division III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.2, 3.5.3 and 3.7.1.1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

*When handling irradiated fuel in the primary containment or Fuel Building.

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3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

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DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized:

a. A.C. power distribution:

1. Division I consisting of:

a) 4160 volt A.C. bus 1ENS*SWG1A
b) 480 volt A.C. bus 1EJS*LDC1A and 1EJS*LDC2A.
c) 480 volt A.C. MCCs 1EHS*MCC 2A, 2C, 2E, 2G, 2J, 2L, 8A, 14A, 15A and 16A.
d) 120 volt A.C. distribution panels 1SCV*PNL2A1, 2A2, 2C1, 2E1, 2G1, 2J1, 2L1, 8A1, 14A1, 15A1, and 16A1 powered by 480 volt MCCs 1EHS*MCC 2A, 2C, 2E, 2G, 2J, 2L, 8A, 14A, 15A and 16A with 1SCM*PNL01A energized from voltage regulating transformer 1SCM*XRC14A1, and 1VBS*PNL01A energized from Uninterruptible Power Supply (UPS) 1ENB*INV01A connected to D.C. Division I 125 volt D.C. bus 1ENB*SWG01A and 480 volt A.C. MCCs 1EHS*MCC8A and 14A.

2. Division II consisting of:

a) 4160 volt A.C. bus 1ENS*SWG1B.
b) 480 volt A.C. bus 1EJS*LDC1B and 1EJS*LDC2B.
c) 480 volt A.C. MCCs 1EHS*MCC 2B, 2D, 2F, 2H, 2K, 8B, 14B, 15B and 16B.
d) 120 volt A.C. distribution panels 1SCV*PNL2B1, 2B2, 2D1, 2F1, 2H1, 2K1, 8B1, 14B1, 15B1 and 16B1 powered by 480 volt MCCs 1EHS*MCC 2B, 2D, 2F, 2H, 2K, 8B, 14B, 15B, and 16B with 1SCM*PNL01B energized from voltage regulating transformer 1SCM*XRC14B1, and 1VBS*PNL01B energized from Uninterruptible Power Supply (UPS) 1ENB*INV01B connected to D.C. Division II 125 volt D.C. Bus 1ENB*SWG01B and 480 volt A.C. MCCs 1EHS*MCC 8B and 14B.

3. Division III consisting of:

a) 4160 volt A.C. bus 1E22*S004
b) 480 volt A.C. MCC 1E22*S002
c) 120 volt A.C. distribution panel 1E22*S002PNL in 480 volt MCC 1E22*S002

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b. D.C. power distribution:

1. Division I, consisting of 125 volt D.C. bus 1ENB*SWG01A, distribution panels 1ENB*PNL02A, 1ENB*PNL03A and 1ENB*PNL04A and MCC 1ENB*MCC1.

2. Division II, consisting of 125 volt D.C. bus 1ENB*SWG01B and distribution panels 1ENB*PNL02B and 1ENB*PNL03B.

3. Division III, consisting of 125 volt D.C. distribution panel 1E22*S001PNL.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. For A.C. power distribution:

1. With either division I or division II of the above required A.C. distribution system not energized, re-energize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With division III of the above required A.C. distribution system not energized, declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.1 and 3.7.1.1.

3. With one of the above required inverters inoperable, energize the associated distribution panel within 8 hours; restore the inoperable inverter to OPERABLE and energized 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.

b. For D.C. power distribution:

1. With either division I or division II of the above required D.C. distribution system not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With division III of the above required D.C. distribution system not energized, declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.1 and 3.7.1.1.

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4.8.3.1 Each of the above required power distribution system divisions shall be determined energized by verifying at least once per 7 days correct breaker alignment on the busses/switchgear/load centers/MCCs/panels. Voltage shall be verified on the following:

Div. I	Div. II	Div. III
1ENS*SWG1A	1ENS*SWG1B	1E22*S004
1EJS*LDC1A	1EJS*LDC1B	1E22*S002
1ENB*SWG01A	1ENB*SWG01B	1E22*S001PNL
1VBS*PNL01A	1VBS*PNL01B	
1EJS*LDC2A	1EJS*LDC2B	

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LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following power distribution system divisions shall be energized:

a. For A.C. power distribution, division I or division II and, when the HPCS system is required to be OPERABLE, division III, with:

1. Division I consisting of:

   a) 4160 volt A.C. bus 1ENS*SWG1A.
   b) 480 volt A.C. bus 1EJS*LDC1A and 1EJS*LDC2A.
   c) 480 volt A.C. MCCs 1EHS*MCC 2A, 2C, 2E, 2G, 2J, 2L, 8A, 14A, 15A and 16A.
   d) 120 volt A.C. distribution panels 1SCV*PNL2A1, 2A2, 2C1, 2E1, 2G1, 2J1, 2L1, 8A1, 14A1, 15A1, and 16A1 powered by 480 volt MCCs 1EHS*MCC 2A, 2C, 2E, 2G, 2J, 2L, 8A, 14A, 15A and 16A with 1SCM*PNL01A energized from voltage regulating transformer 1SCM*XRC14A1, and 1VBS*PNL01A energized from Uninterruptible Power Supply (UPS) 1ENB*INV01A connected to D.C. Division I 125 volt D.C. Bus 1ENB*SWG01A and 480 volt A.C. MCCs 1EHS*MCC 8A and 14A.

2. Division II consisting of:

   a) 4160 volt A.C. bus 1ENS*SWG1B.
   b) 480 volt A.C. bus 1EJS*LDC1B and 1EJS*LDC2B.
   c) 480 volt A.C. MCCs 1EHS*MCC 2B, 2D, 2F, 2H, 2K, 8B, 14B, 15B and 16B.
   d) 120 volt A.C. distribution panels 1SCV*PNL2B1, 2B2, 2D1, 2F1, 2H1, 2K1, 8B1, 14B1, 15B1 and 16B1 powered by 480 volt MCCs 1EHS*MCC 2B, 2D, 2F, 2H, 2K, 8B, 14B, 15B and 16B with 1SCM*PNL01B energized from voltage regulating transformer 1SCM*XRC14B1, and 1VBS*PNL01B energized from Uninterruptible Power Supply (UPS) 1ENB*INV01B connected to D.C. Division II 125 volt D.C. Bus 1ENB*SWG01B and 480 volt A.C. MCCS 1EHS*MCC 8B and 14B.

3. Division III consisting of:

   a) 4160 volt A.C. bus 1E22*S004.
   b) 480 volt A.C. switchgear 1E22*S002.
   c) 120 volt A.C. distribution panel 1E22*S002PNL and 480 volt MCC 1E22*S002.

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b. For D.C. power distribution, division I or division II, and when the HPCS system is required to be OPERABLE, Division III, with:

1. Division I consisting of 125 volt D.C. bus 1ENB*SWG01A and distribution panels 1ENB*PNL02A, 1ENB*PNL03A and 1ENB*PNL04A and MCC 1ENB*MCC1.

2. Division II consisting of 125 volt D.C. bus 1ENB*SWG01B and distribution panels 1ENB*PNL02B and 1ENB*PNL03B.

3. Division III consisting of 125 volt D.C. distribution panel 1E22*S001PNL.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and #.

ACTION:

a. For A.C. power distribution:

1. With less than division I and/or division II of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment or Fuel Building, and operations with a potential for draining the reactor vessel.

2. With division III of the above required A.C. distribution system not energized, declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.2, 3.5.3 and 3.7.1.1.

b. For D.C. power distribution:

1. With less than division I and/or division II of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment or Fuel Building, and operations with a potential for draining the reactor vessel.

2. With division III of the above required D.C. distribution system not energized, declare the HPCS system and the C SSW pump inoperable and take the ACTION required by Specifications 3.5.2, 3.5.3 and 3.7.1.1.

c. The provisions of Specification 3.0.3 are not applicable.

#When handling irradiated fuel in the primary containment or Fuel Building.

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4.8.3.2 At least the above required power distribution system divisions shall be determined energized by verifying at least once per 7 days correct breaker alignment on the busses/switchgear/load centers/MCCs/panels. Voltage shall be verified on the following:

Div. I	Div. II	Div. III
1ENS*SWG1A	1ENS*SWG1B	1E22*S004
1EJS*LDC1A	1EJS*LDC1B	1E22*S002
1ENB*SWG01A	1ENB*SWG01B	1E22*S001PNL
1VBS*PNL01A	1VBS*PNL01B	
1EJS*LDC2A	1EJS*LDC2B	

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 4.16 kV circuit breakers, de-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verifying, at least once per 7 days thereafter, the redundant circuit breaker to be tripped.

   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by racking out the breaker within 72 hours and verifying, at least once per 7 days thereafter, the inoperable breaker(s) to be racked out.

   3. For 480 volt MCC circuit breaker/fuse combination starters, remove the inoperable starter(s) from service by locking the breakers open and removing the control power fuse within 72 hours and verifying, at least once per 7 days thereafter, the inoperable starter(s) circuit breaker to be locked open with the control power fuse removed.

   4. For 120/140 volt molded case circuit breakers, remove the inoperable circuit breaker(s) from service by tripping both 120/140 volt breakers open and locking the upstream 480 volt MCC breaker open within 72 hours and verifying, at least once per 7 days thereafter, the 480 volt MCC breaker(s) to be locked open.

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

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 4.16 kV circuits which have their redundant circuit breakers tripped or to 480 or 120 volt circuits which have the inoperable circuit breaker racked out or locked open.

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4.8.4.1 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage 4.16 kv circuit breakers are OPERABLE by selecting, on a rotating basis, at least one of the four circuit breakers 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 overcurrent control circuits function as designed.

   c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least one of the four circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting currents in excess of the breaker's nominal setpoint and measuring the response time of the long time and short time delay elements and the setpoint of the instantaneous element, as appropriate. The measured data shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of at least 10% of each type of motor starter used for penetration redundant overcurrent protection. Motor starters selected for functional testing shall be selected on a rotating basis. Testing of these motor starters shall consist of injecting a current with a value equal to the locked rotor current of the associated motor and verifying that the motor starter operates to interrupt the current within the associated thermal overload time delay band width for that current as specified by the manufacturer. Motor starters found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each motor starter found inoperable during these functional tests, an additional representative sample of at least 10% of all the motor starters of the inoperable type shall also be functionally tested until no more failures are found or all motor starters 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 program in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.1-1

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PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

A. 4.16 KV Circuit Breakers

PRIMARY PROTECTION		SECONDARY PROTECTION		
LOCATION	DEVICE #	LOCATION	DEVICE #	EQUIPMENT ID#
1. 1ENS*SWG4A	ACB 36	1ENS*SWG3A	ACB 35	1B33-C001A
2. 1ENS*SWG4B	ACB 38	1ENS*SWG3B	ACB 37	1B33-C001B

B. 120/140 VAC Molded Case Circuit Breakers

1. Type Square D

PRIMARY PROTECTION Location	SECONDARY PROTECTION Location	EQUIP. NO.
1LAR-BKR1B	1LAR-BKR1A	1LAR-PNL1R1
1LAR-BKR2B	1LAR-BKR2A	1LAR-PNL1R2
1LAR-BKR3B	1LAR-BKR3A	1LAR-PNL1R3
1LAR-BKR4B	1LAR-BKR4A	1LAR-PNL1R4
1LAR-BKR5B	1LAR-BKR5A	1LAR-PNL1R5
1LAR-BKR6B	1LAR-BKR6A	1LAR-PNL1R6
1LAR-BKR7B	1LAR-BKR7A	1LAR-PNL1R7
1LAR-BKR8B	1LAR-BKR8A	1LAR-PNL1R8
1LAR-BKR9B	1LAR-BKR9A	1LAR-PNL1R9
1LAR-BKR10B	1LAR-BKR10A	1LAR-PNL1R10
1LAR-BKR11B	1LAR-BKR11A	1LAR-PNL1R11
1LAR-BKR12B	1LAR-BKR12A	1LAR-PNL1R12
1LAR-BKR13B	1LAR-BKR13A	1LAR-PNL1R13
1LAR-BKR14B	1LAR-BKR14A	1LAR-PNL1R14
1LAR-BKR16B	1LAR-BKR16A	1LAR-PNL1R16
1LAR-BKR17B	1LAR-BKR17A	1LAR-PNL1R17
1LAR-BKR18B	1LAR-BKR18A	1LAR-PNL1R18
1LAR-BKR19B	1LAR-BKR19A	1LAR-PNL1R19
1SCA-BKR2A12	1SCA-BKR2A11	1SCA-PNL2A1
1SCA-BKR2D12	1SCA-BKR2D11	1SCA-PNL2D1
1SCA-BKR2F12	1SCA-BKR2F11	1SCA-PNL2F2
1SCA-BKR2D14	1SCA-BKR2D13	1SCA-PNL2D3
1SCA-BKR8A22	1SCA-BKR8A21	1SCA-PNL8A2
1SCA-BKR8B22	1SCA-BKR8B21	1SCA-PNL8B2

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

C. 480 VAC Molded Case Circuit Breakers

1. Gould Circuit Breaker Type A821 with Gould Starter/Controller Type FVNR Size 1

Location	Cubicle	Equip. No.
1EHS*MCC2A	2B	1CPM*FN1A
1EHS*MCC2B	2B	1CPM*FN1B
1NHS-MCC2A	2A	1C41-D002
1NHS-MCC2A	3C	1DER-P1A
1NHS-MCC2A	3D	1DER-P2A
1NHS-MCC2A	4D	1DFR-P2A
1NHS-MCC2A	4E	1DFR-P1A
1NHS-MCC2A	6E	1HVR-FN1A
1NHS-MCC2B	4C	1DER-P1B
1NHS-MCC2B	5C	1DER-P2B
1NHS-MCC2B	6B	1DFR-P2B
1NHS-MCC2B	6C	1HVR-FN1D
1NHS-MCC2C	1E	1B33-C001AH
1NHS-MCC2D	3B	1B33-C001BH
1NHS-MCC2E	2B	1G36-C002
1NHS-MCC2E	2C	1HVR-FN1C
1NHS-MCC2E	3B	1G36-C001A
1NHS-MCC2E	4D	1WCS-P5A
1NHS-MCC2E	4E	1B33-D003A2
1NHS-MCC2E	6C	1B33-D003A5
1NHS-MCC2E	1C	1G36-A001AG
1NHS-MCC2F	3B	1G36-C001B
1NHS-MCC2F	3C	1HVR-FN1B
1NHS-MCC2F	4A	1DFR-P1B
1NHS-MCC2F	5A	1WCS-P5B
1NHS-MCC2F	5C	1B33-D003B5
1NHS-MCC2F	6B	1B33-D003B2
1NHS-MCC2F	6C	1G36-A002AG
1NHS-MCC3A	2E	1F42-D002
1NHS-MCC3A	3E	1DFR-P6A
1NHS-MCC3B	3C	1DFR-P6B
1NHS-MCC102B	3A	1CPP-FN1

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TABLE 3.8.4.1-1 (Continued)

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PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTION DEVICES

C. 480 VAC Molded Case Circuit Breakers (Continued)

2. Gould Circuit Breaker Type A822 with Gould Starter/Controller Type FVR Size 1

Location	Cubicle	Equip. No.
1EHS*MCC2A	2A	1C41*MOVF001A
1EHS*MCC2A	5A	1SWP*MOV4A
1EHS*MCC2A	5B	1SWP*MOV5B
1EHS*MCC2A	5C	1SWP*MOV502A
1EHS*MCC2A	6A	1RCS*MOV58A
1EHS*MCC2A	6B	1RCS*MOV59A
1EHS*MCC2A	6C	1SWP*MOV503A
1EHS*MCC2B	1B	1SFC*MOV120
1EHS*MCC2B	1D	1SFC*MOV139
1EHS*MCC2B	2A	1C41*MOVF001B
1EHS*MCC2B	5A	1SWP*MOV4B
1EHS*MCC2B	5B	1SWP*MOV5A
1EHS*MCC2B	5C	1SWP*MOV502B
1EHS*MCC2B	6A	1RCS*MOV58B
1EHS*MCC2B	6B	1RCS*MOV59B
1EHS*MCC2B	6C	1SWP*MOV503B
1EHS*MCC2C	1D	1CCP*MOV142
1EHS*MCC2C	2C	1CCP*MOV143
1EHS*MCC2C	2D	1CPM*MOV1A
1EHS*MCC2C	3A	1CPM*MOV2A
1EHS*MCC2C	3B	1CPM*MOV3A
1EHS*MCC2C	3C	1E12*MOVF037A
1EHS*MCC2C	4A	1E12*MOVF042A
1EHS*MCC2C	4B	1HVN*MOV22A
1EHS*MCC2C	4C	1RCS*MOV60A
1EHS*MCC2C	5B	1RCS*MOV61A
1EHS*MCC2C	5C	1CPM*MOV4A
1EHS*MCC2D	1C	1B21*MOVF016
1EHS*MCC2D	1D	1CPM*MOV1B
1EHS*MCC2D	2C	1CPM*MOV2B
1EHS*MCC2D	2D	1CPM*MOV3B
1EHS*MCC2D	3A	1CPM*MOV4B
1EHS*MCC2D	3B	1CPP*MOV104
1EHS*MCC2D	3C	1E51*MOVF063
1EHS*MCC2D	4A	1E51*MOVF076
1EHS*MCC2D	4B	1G33*MOVF001
1EHS*MCC2D	4C	1G33*MOVF028
1EHS*MCC2D	5A	1WCS*MOV178
1EHS*MCC2K	1D	1CCP*MOV144
1EHS*MCC2K	2A	1RCS*MOV60B
1EHS*MCC2K	2B	1RCS*MOV61B
1EHS*MCC2K	2C	1HVN*MOV22B

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

C. 480 VAC Molded Case Circuit Breakers (Continued)

2. Gould Circuit Breaker Type A822 with Gould Starter/Controller Type FVR Size 1 (Continued)

Location	Cubicle	Equip. No.
1EHS*MCC2K	3D	1E12*MOVF042B
1EHS*MCC2K	4A	1E12*MOVF009
1EHS*MCC2K	4D	1G33*MOVF053
1EHS*MCC2K	5A	1G33*MOVF040
1EHS*MCC2K	6C	1HVN*MOV102
1EHS*MCC2K	6D	1E12*MOVF037B
1EHS*MCC2K	7D	1CCP*MOV158
1NHS-MCC2A	1C	1B21-MOVF001
1NHS-MCC2A	1D	1B33-MOVF023A
1NHS-MCC2A	5C	1G33-MOVF102
1NHS-MCC2A	5D	1B33-MOVF067A
1NHS-MCC2A	7D	1G33-MOVF106
1NHS-MCC2B	3B	1G33-MOVF042
1NHS-MCC2B	3C	1B21-MOVF002
1NHS-MCC2B	4D	1G33-MOVF044
1NHS-MCC2B	5D	1G33-MOVF100
1NHS-MCC2B	6D	1G33-MOVF101
1NHS-MCC2D	2E	1B21-MOVF005
1NHS-MCC2D	3D	1B33-MOVF067B
1NHS-MCC2D	4D	1B33-MOVF023B
1NHS-MCC2E	3A	1G33-MOVF031
1NHS-MCC2E	5E	1G33-MOVF107
1NHS-MCC2F	2D	1G33-MOVF104
1NHS-MCC8A	4E	1C11-MOVF003

3. Gould Circuit Breaker Type HE43

1NHS-MCC2A	2C	1POP-WR2A01
1NHS-MCC2A	2D	1POP-WR2A02
1NHS-MCC2C	1CT	1H22-PNLP008
1NHS-MCC2D	5C	1POP-WR2D01
1NHS-MCC2D	5D	1POP-WR2D02
1NHS-MCC8A	1E	1F15-E006
1NHS-MCC8A	2D	1F15-E005
1NHS-MCC8A	4C	1F11-E012
1NHS-MCC8A	6B	1FNR-P06
1NHS-MCC8A	6C	1FNR-P08
1NHS-MCC8B	2A	1FNR-P07
1NHS-MCC2F	2A	1POP-WR2F01
1NHS-MCC2F	2B	1JRB-EL1A
1NHS-MCC2E	3C	1MHR-CRN2
1NHS-MCC2A	3A	1FNR-P09
1NHS-MCC2A	4A	1FNR-P10
1NHS-MCC2B	1C	1FNR-P11
1NHS-MCC8A	3D	1MHR-CRN3

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TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

C. 480 VAC Molded Case Circuit Breakers (Continued)

4. Gould Circuit Breaker Type A80 with Gould Starter/Controller Type FVNR Size 3

Location	Cubicle	Equip. No.
1EHS*MCC2A	2C	1C41*C001A
1EHS*MCC2B	2C	1C41*C001B
1NHS-MCC2B	2D	1C41*D003
1NHS-MCC2E	1D	1B33-D003A1
1NHS-MCC2E	6D	1B33-D003A4
1NHS-MCC2F	4D	1B33-D003B1
1NHS-MCC2F	6D	1B33-D003B4

5. Gould Circuit Breaker Type A80 with Gould Starter/Controller Type 2SP1W Size 4

1NHS-MCC102A	1C	1DRS-UC1A
1NHS-MCC102A	2C	1DRS-UC1C
1NHS-MCC102A	3B	1DRS-UC1E
1NHS-MCC102B	1C	1DRS-UC1B
1NHS-MCC102B	2C	1DRS-UC1D
1NHS-MCC102B	3B	1DRS-UC1F

6. Gould Circuit Breaker with Type A821 Gould Starter/Controller Type FVNR Size 2

1NHS-MCC8B	1D	1F42-E001

D. Air Circuit Breakers - GE Type ARR

Location	Device No.	Location	Device No.	Equip. No.
1EJS*LDC2B	ACB79	1EJS*LDC2B	ACB78	1HVR-UC1C
1EJS*LDC2A	ACB36	1EJS*LDC2A	ACB38	1HVR*UC1A
1EJS*LDC2A	ACB22	1EJS*LDC2A	ACB38	1MHR*RN1C
1EJS*LDC2B	ACB76	1EJS*LDC2B	ACB78	1HVR*UC1B
1EJS*LDC2A	ACB23	1HCS*PWRS1A	Int. Fuse	1HCS*RBNR1A
1EJS*LDC2B	ACB63	1HCS*PWRS1B	Int. Fuse	1HCS*RBNR1B

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OTHER OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The overcurrent protection devices shown in Table 3.8.4.2-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With one or more of the overcurrent protective devices shown in Table 3.8.4.2-1 inoperable, remove the inoperable circuit breaker(s) from service by racking out the breaker within 72 hours and return the breaker(s) to OPERABLE status within 7 days. Otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.2 The overcurrent protective devices shall be demonstrated OPERABLE at least once per 18 months by selecting and testing one-half of each type of circuit breaker on a rotating basis. Testing of these circuit breakers shall consist of injecting currents in excess of the breaker's nominal setpoint and measuring the response time of the long time and short time delay elements and the setpoint of the instantaneous element, as appropriate. The measured data shall be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer.

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TABLE 3.8.4.2-1

OTHER OVERCURRENT PROTECTIVE DEVICES

TYPE

1. Main Control Room Lighting

   Protective Device

   1EHS*MCC14A
   1EHS*MCC14B

2. RPS Alternate Source of Power

Primary Protection	Secondary Protection
1EHS*MCC14A	1RPS*XRC10A
1EHS*MCC14B	1RPS*XRC10B

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REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two RPS electric power monitoring channels for each in-service RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an in-service RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an in-service RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the unit is in COLD SHUTDOWN for a period of more than 24 hours, unless performed within the previous six months, and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Over-voltage $\leq$ 132 VAC, Bus A and B.

   2. Under-voltage $\geq$ 115 VAC, Bus A and B, and

   3. Under-frequency 57 Hz, + 2, - 0%, Bus A and B.

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A.C. CIRCUITS INSIDE CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.4 At least the following A.C. circuits inside containment shall be de-energized*:

Equipment ID	Location	Device
1MHR*CRN1	1EJS*LDC2A	ACB022
1F42-PNLP003	1SCA-PNL8C1	Circuit Breaker 1
1F42-D002H	1SCA-PNL8C1	Circuit Breaker 15
1SFT-PNL106	1SCA-PNL8B2	Circuit Breaker 2
1SFT-PNL106	1SCA-PNL8B2	Circuit Breaker 10
1HVR*UC1AH	1SCV*PNL2A2	Circuit Breaker 5
1HVR*UC1BH	1SCV*PNL2B2	Circuit Breaker 12
1HVR-UC1CH	1SCA-PNL2C1	Circuit Breaker 9
1HVR-FN1AH	1SCA-PNL2A2	Circuit Breaker 3
1HVR-FN1BH	1SCA-PNL2F1	Circuit Breaker 6
1HVR-FN1CH	1SCA-PNL2E1	Circuit Breaker 1
1HVR-FN1DH	1SCA-PNL2B1	Circuit Breaker 6
1DRS-UC1AH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1BH	1SCA-PNL2F1	Circuit Breaker 3
1DRS-UC1CH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1DH	1SCA-PNL2F1	Circuit Breaker 3
1DRS-UC1EH	1SCA-PNL2E1	Circuit Breaker 2
1DRS-UC1FH	1SCA-PNL2F1	Circuit Breaker 3
1WCS-P5AH	1SCA-PNL2E1	Circuit Breaker 4
1WCS-P5BH	1SCA-PNL2F1	Circuit Breaker 2

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 location within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.4 Each of the above required A.C. circuits shall be determined to be de-energized by verifying at least once per 24 hours** that the associated circuit breakers are in the tripped condition.

*Except during entry into the containment.

**Except at least once per 31 days if locked, sealed or otherwise secured in the tripped condition.

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3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be 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 at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform main hoist fuel-loaded.

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.

*See Special Test Exceptions 3.10.1 and 3.10.3.
#The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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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 by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

*The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

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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 with:

a. Continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. Unless adequate shutdown margin has been demonstrated, either the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn# or the rod pattern control system shall be OPERABLE per Specification 3.1.4.2.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

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.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

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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 0.7 cps*:

   1. Prior to control rod withdrawal,

   2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours, except that:

      a. During fuel unloading, the required count rate may be permitted to be less than 0.7 cps*.

      b. Prior to and during fuel loading, until sufficient fuel has been loaded to maintain at least 0.7 cps*, the required count rate may be achieved by:

         a) Use of portable external source, or

         b) Loading up to 2 fuel assemblies in cells containing inserted control rods around an SRM.

d. Verifying within 8 hours prior to and at least once per 12 hours during:

   1. The time any control rod is withdrawn,## or

   2. Shutdown margin demonstrations,

   that the RPS circuitry "shorting links" have been removed.

---

*Provided signal to noise ratio ≥ 2, otherwise use 3.0 cps.

##Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be 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 (1) control rods removed per Specification 3.9.10.1 or 3.9.10.2 or (2) one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

**See Special Test Exception 3.10.3.

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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.

SURVEILLANCE REQUIREMENTS

4.9.4 The reactor shall be determined to have been subcritical for at least 24 hours by verification, prior to movement of irradiated fuel in the reactor pressure vessel, of the date and time of subcriticality.

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3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the main control room and refueling platform personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the main control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the main 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.

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3/4.9.6 REFUELING AND/OR FUEL HANDLING PLATFORM

REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.1 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods.

APPLICABILITY: During handling of fuel assemblies or control rods.

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 after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each refueling platform hoist used for handling of control rods or fuel assemblies shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist before the load exceeds 1200 ± 50 pounds.

b. Demonstrating operation of the overload cutoff on the frame-mounted and monorail-mounted auxiliary hoists when the load exceeds 500 ± 50 pounds.

c. Demonstrating operation of the uptravel interlock when uptravel brings the top of the active fuel or control rod to 8 feet, 6 inches below the water level.

d. Demonstrating operation of the downtravel interlock on the main hoist when grapple hook down travel reaches 4 inches below fuel assembly handle during operations within the reactor pressure vessel.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds 485 ± 50 pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 ± 50 pounds.

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3/4.9.6 REFUELING AND/OR FUEL HANDLING PLATFORM

FUEL HANDLING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6.2 The fuel handling platform shall be OPERABLE and used for handling fuel assemblies or control rods.

APPLICABILITY: During handling of fuel assemblies or control rods.

ACTION:

With the requirements for fuel handling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6.2 Each fuel handling platform hoist used for handling of control rods or fuel assemblies shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist before the load exceeds 1100 ± 50 pounds.

b. Demonstrating operation of the overload cutoff on the monorail-mounted auxiliary hoists when handling control rods before the load exceeds 500 ± 50 pounds, and when handling unirradiated fuel before the load exceeds 1000 ± 50 pounds.

c. Demonstrating operation of the uptravel interlock when uptravel brings the top of the active irradiated fuel or control rods to 8 feet, 6 inches below the water level.

d. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

e. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds 350 ± 50 pounds.

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3/4.9.7 CRANE TRAVEL - SPENT AND NEW FUEL STORAGE, TRANSFER AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent or new fuel storage, transfer or upper containment fuel pool racks.

APPLICABILITY: With fuel assemblies in the spent or new fuel storage, transfer or upper containment fuel pools.

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

4.9.7.1 The fuel building crane loads shall be verified to weigh less than or equal to 1200 pounds before travel over fuel assemblies in the spent or new fuel storage pools and the lower transfer pools.

4.9.7.2 The reactor building polar crane loads shall be verified to weigh less than or equal to 1200 pounds before travel over fuel assemblies in the upper transfer and containment fuel pools.

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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 the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel, while in OPERATIONAL CONDITION 5, when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

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.

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.

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3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

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 and upper containment fuel pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage or upper containment fuel pools.

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 or upper containment fuel pool areas, as applicable, 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 At least once per 7 days, the water level in the spent fuel storage and upper containment fuel pools shall be determined to be at least at its minimum required depth.

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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 core and/or 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 position or in the Refuel position per Table 1.2 and 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 inserted and electrically or hydraulically disarmed, or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITION 4 and 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 core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

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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 core and/or 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 inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the 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 per Specification 3.9.10.1.c.

d. All other control rods, in a five-by-five array centered on the control rod being removed, are inserted and electrically or hydraulically disarmed, or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

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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 core and/or 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 inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the 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 inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended unless all control rods are inserted in the core.

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 core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

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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 core and/or 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 inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the 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 inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations are suspended unless all control rods are inserted in 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.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 23 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish PRIMARY CONTAINMENT INTEGRITY - FUEL HANDLING within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method, and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified, at least once per 12 hours, to be in operation and circulating reactor coolant.

*The shutdown cooling loop may be removed from operation for up to 2 hours per 8-hour period.

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each train consisting of at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 23 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode train.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method, and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified, at least once per 12 hours, to be in operation and circulating reactor coolant.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

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3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The floor plugs are installed and the access door of all rooms through which the transfer system penetrates are closed and locked.

b. All access interlocks and palm switches are OPERABLE.

c. The blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. At least one IFTS carriage position indicator at each carriage position is OPERABLE and at least one liquid level sensor is OPERABLE.

e. The keylock switch which provides access control lockout is OPERABLE.

f. The warning lights outside of the access doors are OPERABLE.

APPLICABILITY: When the IFTS containment blank flange is removed.

ACTION:

a. With one or more access interlocks, warning lights, and/or palm switches inoperable, operation of the IFTS may continue provided that entry into the area is prohibited by establishing a continuous watch and conspicuously posting as a high radiation area.

b. With the requirements of the above specification not otherwise satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12.1 Within 1 hour prior to the startup of the IFTS, verify that no personnel are in areas immediately adjacent to the IFTS tube and that the floor plugs are installed and access doors, to rooms through which the IFTS tube penetrates, are closed and locked.

4.9.12.2 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, verify that:

a. At least one IFTS carriage position indicator at each carriage position is OPERABLE and at least one level sensor is OPERABLE.

b. The warning lights outside of each access door are OPERABLE and the floor plug is installed.

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SURVEILLANCE REQUIREMENTS

4.9.12.3 Within 4 hours prior to the operation of IFTS and at least once per 7 days thereafter, verify that:

a. The access interlock and palm switch are OPERABLE for the containment isolation valve room.

b. The blocking valve in the Fuel Building IFTS hydraulic power unit is OPERABLE.

c. The keylock switch which provides access control lockout is OPERABLE.

4.9.12.4 Within 4 hours prior to installation of the floor plugs, after they have been removed, verify that the access interlocks and palm switches for the Fuel Building and Shield Building Annulus IFTS support rooms are OPERABLE.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY/DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.2, 3.6.1.4, 3.6.2.1, 3.6.2.3 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 and the primary containment and drywell air lock doors to be open when the reactor mode switch is 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 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°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, at least once per hour during low power PHYSICS TESTS, to be within the limits.

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3/4.10.2 ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod pattern control system (RPCS) per Specification 3.1.4.2 may be suspended, by means of the individual rod position bypass switches, for the following tests:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

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 RPCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RPCS are bypassed, verify:

a. Within 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed, that movement of the control rods from 75% ROD DENSITY to the RPCS low power setpoint is limited to the established control rod sequence for the specified test, and

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

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3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specifications 3.9.1 and 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 at least the following requirements are satisfied.

a. The source range monitors are OPERABLE per Specification 3.9.2 with:

   1. The RPS circuitry "shorting links" removed, or

   2. The rod pattern control system OPERABLE per Specification 3.1.4.2.

b. Conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "continuous withdrawal" control shall not be used during out-of-sequence movement of the control rods.

d. 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 and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2, with:

   1. The RPS circuitry "shorting links" removed, or

   2. The rod pattern control system OPERABLE.

b. A second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

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3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3, that recirculation loops be in operation with matched flow, may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

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, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, 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, at least once per hour during PHYSICS TESTS and the Startup Test Program, to be less than 24 hours.

4.10.4.2 THERMAL POWER shall be determined, at least once per hour during PHYSICS TESTS, to be less than 5% of RATED THERMAL POWER.

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3/4.10.5 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.5.1 may be suspended, during training startups, to permit one RHR subsystem to be aligned in the shutdown cooling mode, 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 200°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

4.10.5 At least once per hour during training startups, the reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature to be within the limits.

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

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CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1.1.1-1

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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TABLE 4.11.1.1.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Batch Waste Release (Liquid Radwaste Recovery Sample Tanks[b])	P Each Batch	P Each Batch	Principal Gamma Emitters[c]; except for Ce-144	5×10^{-7} 5×10^{-6}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE NOTATION

a - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

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TABLE 4.11.1.1.1-1 (continued)

TABLE NOTATION (continued)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample, as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and the time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

b - A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c - The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

d - A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose, from the release of radioactive materials in liquid effluents, exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined, at least once per 31 days, in accordance with the methodology and parameters in the ODCM.

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LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses, due to the liquid effluent, to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

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LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any unprotected outdoor tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above unprotected outdoor tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank; within 48 hours reduce the tank contents to within the limit; and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above unprotected outdoor tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

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3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate, due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1), shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate, due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents, shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11.2.1.2-1.

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TABLE 4.11.2.1.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Main Plant Exhaust Duct	M[c] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
B. Fuel Building Ventilation Exhaust Duct	M[d] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. Radwaste Building Ventilation Exhaust Duct	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
D. All Release Types as listed in A, B, C above.	Continuous[e]	W[f] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[e]	W[f] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[e]	Q Composite Particulate Sample	SR-89, SR-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATION

a - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size, in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

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TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATION

b - The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

c - Sampling and analysis shall also be performed, within one hour following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER, unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3 and (2) the condenser offgas noble gas activity monitor shows that offgas activity has not increased by more than a factor of 3.

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

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

f - Samples shall be changed at least once per 7 days, and analyses shall be completed within 48 hours after changing or after removal from sampler. Sampling shall also be performed at least once per 24 hours for the main plant exhaust for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the condenser offgas noble gas monitor shows that offgas activity has not increased more than a factor of 3.

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DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 At least once per 31 days, cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM.

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DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents released to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1), shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release, in gaseous effluents, of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 At least once per 31 days, cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, shall be determined in accordance with the methodology and parameters in the ODCM.

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GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM inoperable for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 The instruments specified in the ODCM shall be checked every 12 hours, whenever the main condenser air ejector is in operation, to ensure that the GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM is functioning.

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VENTILATION EXHAUST TREATMENT

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LIMITING CONDITION FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses, due to gaseous effluent releases to areas at and beyond the SITE BOUNDARY (see Figures 5.1.1-1 and 5.1.3-1), would exceed 0.3 mrem to any organ in a 31 day period.

APPLICABILITY: At all times other than when the VENTILATION EXHAUST TREATMENT system is undergoing routine maintenance.

ACTION:

a. With gaseous waste being discharged from the ventilation exhaust ducts without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 Doses due to gaseous releases from the site shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

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EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser offgas treatment system is in operation.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits by continuously monitoring, with the hydrogen monitor(s) required in compliance with Specification 3.3.7.11, the waste gases in the main condenser offgas treatment system whenever the main condenser evacuation system is in operation.

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LIMITING CONDITION FOR OPERATION

3.11.2.7 The release rate of the sum of the activities from the noble gases* measured prior to the holdup pipe shall be limited to less than or equal to 290 millicuries/sec after 30 minutes decay.

APPLICABILITY: Whenever the main condenser offgas treatment system is in operation.

ACTION:

With the release rate of the sum of the activities from the noble gases* prior to the holdup pipe exceeding 290 millicuries/sec after 30 minutes decay, restore release rate to within its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity of noble gases prior to the holdup pipe shall be continuously monitored in accordance with Specification 3.3.7.1.

4.11.2.7.2 The release rate of the sum of the activities from the noble gases* measured prior to the holdup pipe shall be determined, at the following frequencies, to be within the limits of Specification 3.11.2.7 by performing an isotopic analysis of a representative sample of gases taken prior to the holdup pipe.

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

*Gamma scintillation detectors are used to measure the Kr-85m, -87, -88 and Xe-133, -133m, -135, -138 contribution after 30 minutes decay.

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3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

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

SURVEILLANCE REQUIREMENTS

4.11.3 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and to radiation from uranium fuel cycle sources, shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made, including direct radiation contributions from the reactor units and from outside storage tanks, to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and that includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits and, if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3 and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor unit and from any unprotected outdoor storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4, Action a.

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LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity, as the result of plant effluents, in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant affluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or broad leaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12.1-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semi-annual Radioactive Effluent Release Report and include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12.1-1 and the detection capabilities required by Table 4.12.1-1.

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TABLE 3.12.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	40 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY; an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site; the balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: 3 samples from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average groundlevel D/Q.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following

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TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
	1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q. 1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]		filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
3. WATERBORNE			
a. Surface[f]	1 sample upstream and 1 sample downstream.	Weekly samples composited monthly or quarterly.	Gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
	Discharge line.	Composite sample over 1-month period[g]	
b. Ground	Samples from 1 or 2 sources only if likely to be affected[h]	Quarterly	Gamma isotopic[e] and tritium analysis quarterly.
c. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic[e] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

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TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
a. Milk (cont'd)	none, then 1 sample from milking animals in each of 3 areas 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.[i] 1 sample from milking animals at a control location 15-30 km distant and in the least prevalent wind direction.		
b. Fish and Invertebrates	1 sample of each of three commercially and/or recreationally important species in vicinity of plant discharge area. 1 sample of each of three species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis[e] on edible portions.
c. Food Products	Samples of 3 different kinds of broad leaf vegetation grown near each of two different locations near the site boundary of highest predicted annual average ground level D/Q if milk sampling is not performed.	Monthly during the growing season.	Gamma isotopic[e] and I-131 analysis.

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TABLE 3.12.1-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Continued	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant near the least prevalent wind direction, if milk sampling is not performed.	Monthly during the growing season.	Gamma isotopic[e] and I-131 analysis.

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TABLE 3.12.1-1 (Continued)

TABLE NOTATION

a - The ODCM shall include, in a table and figures, specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, for each sample location in Table 3.12.1-1. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples, and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

b - One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

c - The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

d - Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12.1-1 (Continued)

TABLE NOTATION (Continued)

e - Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

f - The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence.

g - Composite samples shall be collected weekly up to October 1, 1985. Thereafter, samples shall be collected at intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

h - Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

i - The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

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TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	Reporting Levels				
Analysis	Water[(a)] (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Nb-95	400				
Zr-95	400				
I-131	2**	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			300	
La-140	200			300	

(a)For discharge line samples, these values may be increased by a factor of 11.4 to account for near-field dilution by the Mississippi River.

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/ℓ may be used.

**If no drinking water pathway exists, a value of 20 pCi/ℓ may be used.

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TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b,c]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Nb-95	15					
Zr-95	30					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
La-140	15			15		
Ba-140	60			60		

*If no drinking water pathway exists, a value of 3000 pCi/ℓ may be used.

**If no drinking water pathway exists, a value of 15 pCi/ℓ may be used.

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TABLE NOTATION

a - This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

b - Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

c - The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size, in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

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TABLE 4.12.1-1 (Continued)

TABLE NOTATION (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location, in each of the 16 meteorological sectors, of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

b. With land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 At least once per 12 months, the land use census shall be conducted during the growing season, using that information that will provide the best results, such as by a door-to-door survey or aerial survey or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

*In lieu of the garden census, broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs. Specifications for broad leaf vegetation sampling in Table 3.12.1-1, 4c shall be followed, including analysis of control samples.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials, that correspond to samples required by Table 3.12.1-1, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report to the Commission, in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7, the corrective actions taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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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 measures to be taken for those 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 requires two main control room air conditioning subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, measures must be initiated within 1 hour to place the unit in at least STARTUP within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 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.

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4.0.1 This specification provides that surveillance activities, necessary to ensure the Limiting Conditions for Operation are met, will be performed during the OPERATIONAL CONDITIONS or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities, to be performed without regard to the applicable OPERATIONAL CONDITIONS or other conditions, are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance; instead, it permits the more frequent performance of surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these 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 Condition 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.

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 ne form-ing the inservice inspection and testing activities required by Sectic of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This arifi-cation 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 code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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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/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/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 in-sequence 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 required 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 any time 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 comparison of 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.

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3/4.1.3 CONTROL RODS

The specifications of this section (1) ensure that the minimum SHUTDOWN MARGIN is maintained and the control rod insertion times are consistent with those used in the safety analyses, and (2) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but 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 that is long enough to permit determining the cause of the inoperability yet 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 not fully inserted 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 shut down 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 MCPR from becoming less than 1.06 during the limiting power transient analyzed in Section 15.0 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 MCPR remains greater than 1.06. 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.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

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 reactor.

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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 that could have affected the control rod coupling integrity. 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 ROD PATTERN CONTROL SYSTEM

The rod withdrawal limiter system input power signal originates from the first stage turbine pressure. When operating with the steam bypass valves open, this signal indicates a core power level which is less than the true core power. Consequently, near the low power setpoint and high power setpoint of the rod pattern control system, the potential exists for non-conservative control rod withdrawals. Therefore, when operating at a sufficiently high power level, there is a small probability of violating fuel Safety Limits during a licensing-basis rod withdrawal error transient. To ensure that fuel Safety Limits are not violated, this specification prohibits control rod withdrawal when a biased power signal exists and core power exceeds the specified level.

Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence 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. Therefore, requiring the RPCS to be OPERABLE, when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER, provides adequate control.

ROD PATTERN CONTROL SYSTEM (Continued)

The RPCS provides 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 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report[1] and two supplements[2, 3].

The RPCS is also designed to automatically prevent fuel damage, during higher power operation, in the event of erroneous rod withdrawal from locations of high power density.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. This concentration is increased by 25% to allow for potential leakage and imperfect mixing. The required concentration is achieved by having a minimum available quantity of 3542 gallons of sodium pentaborate solution containing a minimum of 4246 pounds of sodium pentaborate. This quantity of solution is a net amount which is above the pump suction, thus allowing for the portion that cannot be injected. The pumping rate of 41.2 gallons per minute (gpm) per pump provides a negative reactivity insertion rate, over the permissible pentaborate solution volume range, which adequately compensates for the positive reactivity effects due to temperature and xenon decay during shutdown. The temperature versus concentration requirement is necessary to ensure that the sodium pentaborate remains in solution.

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.

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1. C. J. Paone, R. C. Stirn and J. A. Woolley, "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 NEDO-10527, July 1972
3. J. M. Haun, C. J. Paone and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973

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STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established 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. Therefore, 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 due to deterioration of the charges.

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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 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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, 3.2.1-3, 3.2.1-4 and 3.2.1-5.

The daily requirement for calculating APLHGR 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 to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape that could place operation into a condition exceeding a thermal limit.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5 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 NEDE-20566[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.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

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 power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram trip setpoint and the flow biased neutron flux-upscale control rod block trip setpoints of the APRM instruments must be adjusted to ensure that the MCPR does not become less than 1.06 or that $\geq$ 1% plastic strain does not occur 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 THERMAL POWER and CMFLPD indicates a peak power distribution, to ensure that an LHGR transient would not be increased in degraded conditions.

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Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... 3015 Mwt* which corresponds to 105% of rated steam flow

Vessel Steam Output .... 13.08×10^6 lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure.............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks $2.2\ ft^2$.

b. Small Breaks $0.09\ ft^2$.

Fuel Parameters:

FUEL TYPE	FUEL ASSEMBLY 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	1.17

A more detailed listing of input of each model and its source is presented in Section II of NEDE 20566(1) and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes an assembly power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

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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 1.06 and an analysis of abnormal operational transients. For any abnormal operating transient analysis, 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 settings 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 1.06, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and is presented in Figure 3.2.3-1. The power-flow map of Figure B 3/4 2.3-1 shows typical regions of plant operation.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a GE core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[(3)] and the program used in non-pressurization events is described in NEDO-10802[(2)]. 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 TASC code described in NEDE-25149[(4)]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s 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 calculated MCPR at a given point of core flow is defined as $MCPR_f$.

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MINIMUM CRITICAL POWER RATIO (Continued)

The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including localized events such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level.

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 these low power levels, 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 within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape that could place operation into a condition exceeding 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 rate even if fuel pellet densification is postulated.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.

2. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, NEDO-10802, February 1973.

3. Qualification of the One Dimensional Core Transient Model For Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program For The Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

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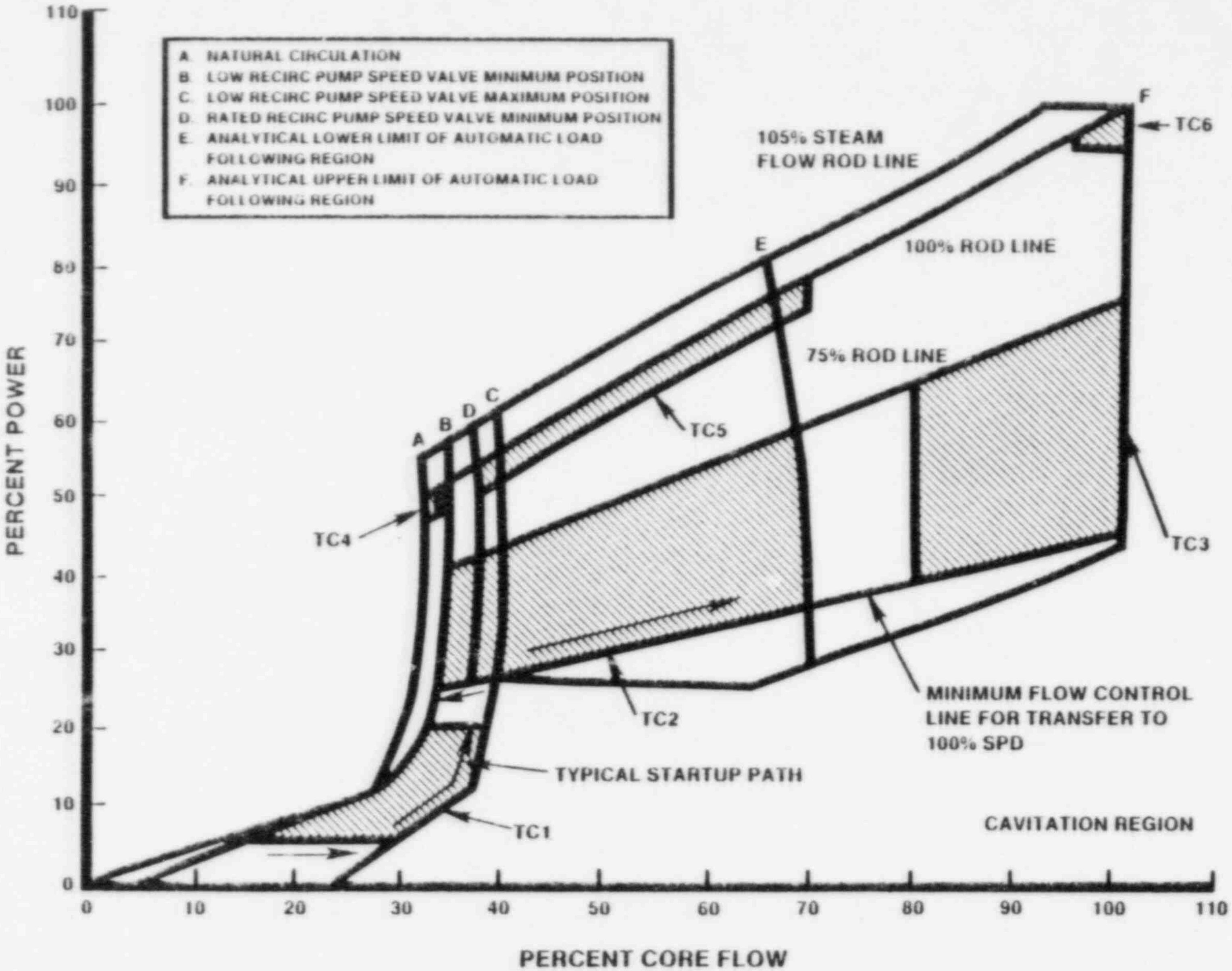

BASES FIGURE B3/4.2.3-1
POWER FLOW OPERATING MAP

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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 four logic channels. The logic channels A(A1) and C(A2) comprise one trip system and the logic channels B(B1) and D(B2) comprise the other trip system for determining compliance with technical specifications. Placement of either logic channel of a trip system in the tripped condition places the trip system in the tripped condition. The trip systems as defined above are independent of each other. There are usually four instrument channels (one in each logic channel) to monitor each parameter. The tripping of a logic channel in each trip system will result in a reactor scram.

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 safety analyses. 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.

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 requirements, 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 has 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 10 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 10 second diesel startup. The safety analysis considers an allowable reactor coolant inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 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 part of the ISOLATION SYSTEM RESPONSE TIME.

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.

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.

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.

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 Reports NEDO-10349, dated March 1971, and NEDO-24222, dated December 1979, and Section 15.8 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 40% of RATED THERMAL POWER, are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the breaker electric arc, i.e., 140 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression and the response time of the system logic.

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.

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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.

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.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Rod Pattern Control System, Section 3/4.2, Power Distribution Limits and Section 3/4.3, Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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.

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 continuously measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63 and 64.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to permit prompt determination of the magnitude of a seismic event and evaluation of 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.

MONITORING INSTRUMENTATION (Continued)

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 permit evaluation of 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 maintenance 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.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to permit monitoring and assessment of 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 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information about the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall 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 the SRMs 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.

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MONITORING INSTRUMENTATION (Continued)

3/4.3.7.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures both that adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and are integral elements in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors for fire suppression must be greater than the minimum number of detectors for fire warning.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.9 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or lessen damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

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MONITORING INSTRUMENTATION (Continued)

3/4.3.7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. In addition, the radioactive release paths of the Fuel Building Ventilation Exhaust, Main Plant Exhaust Duct, and the Radwast Building Ventilation Exhaust include post-accident monitors.

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.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided to initiate action of the containment ventilation system and the feedwater system/main turbine trip system. The containment ventilation system provides emergency containment heat removal as described in Bases 3/4.6.3. The feedwater system/main turbine trip system is initiated in the event of failure of the feedwater controller under maximum demand.

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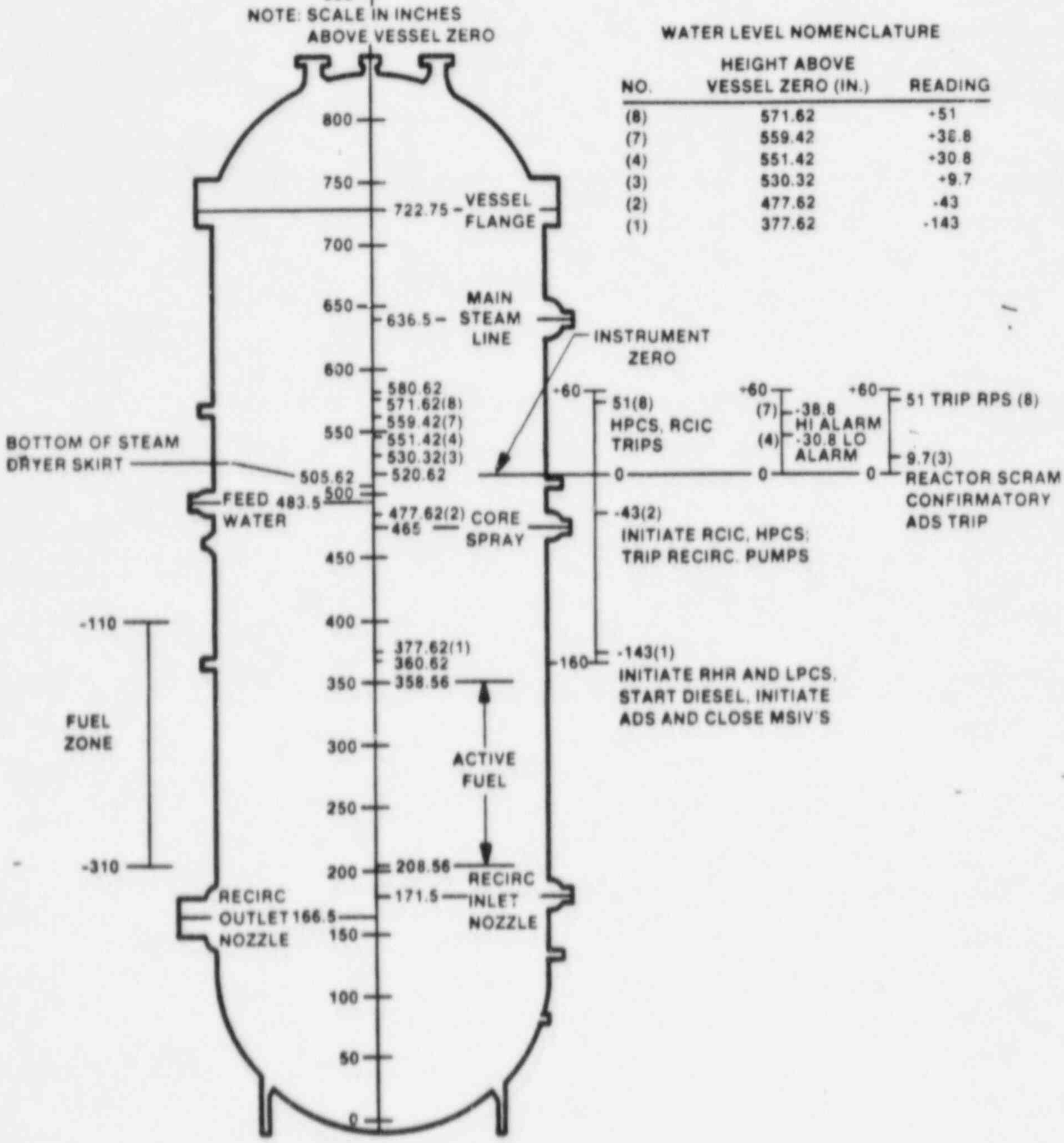

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (IN.)	READING
(8)	571.62	+51
(7)	559.42	+38.8
(4)	551.42	+30.8
(3)	530.32	+9.7
(2)	477.62	-43
(1)	377.62	-143

Bases Figure B 3/4.3-1

REACTOR VESSEL WATER LEVEL

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, and such operation has been determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design basis accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. Recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

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 coolant 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 safety valve function of the safety/relief valves (SRV) is to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig, in accordance with the ASME Code. A total of 9 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 4 SRVs operating in the relief mode and 5 SRVs operating in the safety mode is acceptable.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

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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. In conformance with Regulatory Guide 1.45, the atmospheric gaseous radioactivity system will have a sensitivity of 10^{-6} μCi/cc.

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 shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant 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 0.2 ppm 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 a 0.5 ppm concentration of chlorides is 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.

3/4.4.4 CHEMISTRY (Continued)

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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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, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods, with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomena 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.

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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 3.9 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 thermally 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 thermally induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress-controlling location, each heatup rate of interest must be analyzed on an individual basis.

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 Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." 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.

The actual shift in RT_{NDT} of the vessel material will be determined periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area.

PRESSURE/TEMPERATURE LIMITS (Continued)

The irradiated specimens 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 specimen data and recommendations of Regulatory Guide 1.99, Revision 1.

The pressure/temperature limit lines shown in Figure 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. However, single failure considerations require that two valves be OPERABLE. 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. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

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 1977 Edition and Addenda through Summer 1978.

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).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and adequate cooling mixing to assure accurate temperature indication. However, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	P(%)	HIGHEST STARTING RT_{NDT}(°F)	MAX. * RT_{NDT}(°F)	AVG. UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	C3138-2	0.08	0.012	+9	48	79	+57
Weld	SHELL COURSE No.2 Vertical Seam 3	492L4871/ A421B27AF	0.03	0.020	-50	80	130	+30

NOTE:* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT}(°F)
Shell Ring	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Flange	SA 508 Cl.2	ALL HEATS	+10
Vessel Flange	SA 508 Cl.2	ALL HEATS	+10
Feedwater Nozzle	SA 508 Cl.2	ALL HEATS	-20
Weld	LOW ALLOY STEEL	ALL HEATS	-20
Closure Studs	SA 540 GRADE B23 or B24	ALL HEATS	Meets requirement of 45 ft-lbs and 25 mils lateral expansion at + 10°F

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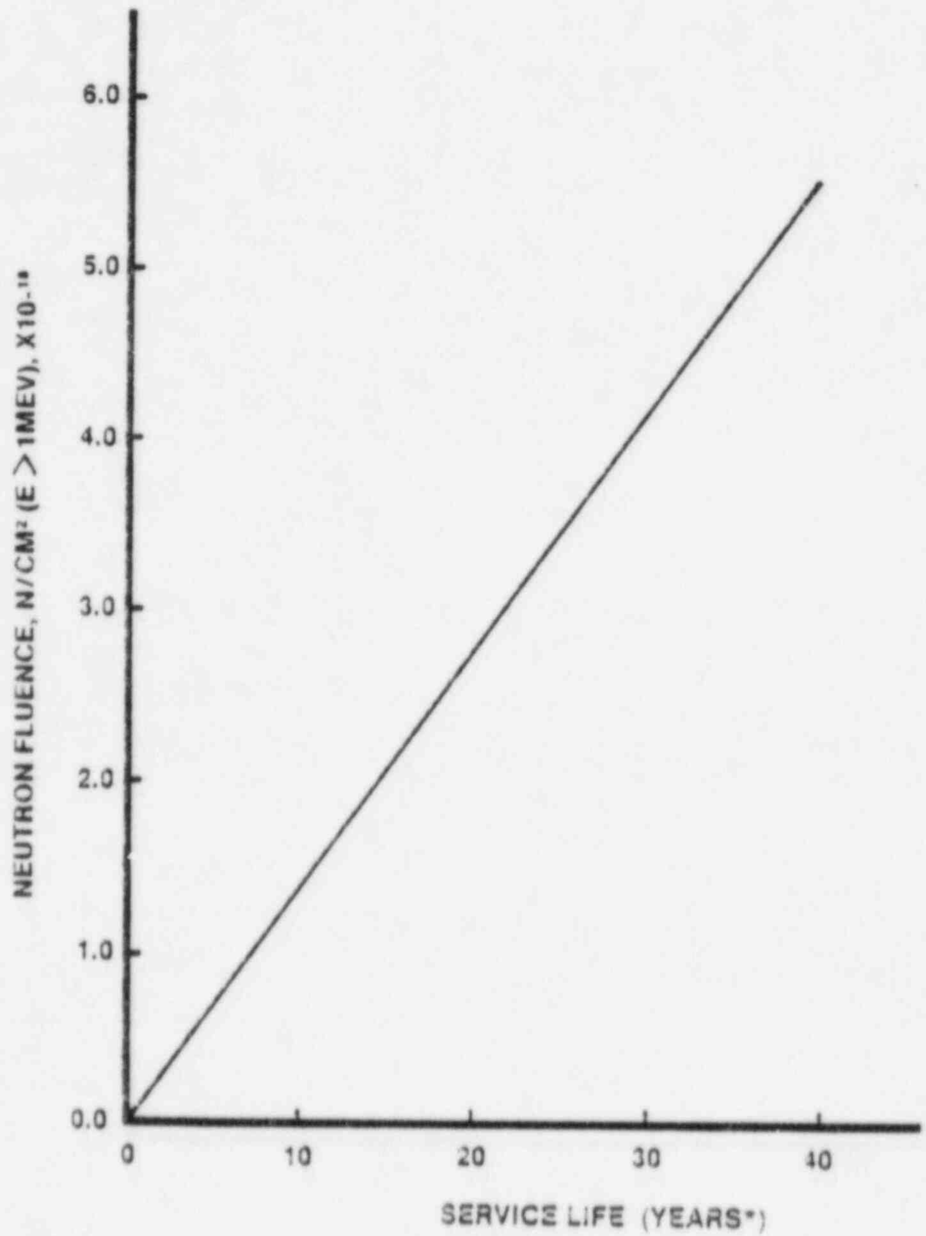

BASES FIGURE B 3/4.4.6-1
FAST NEUTRON FLUENCE (E >1 MEV) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

*AT 90% OF RATED THERMAL POWER AND 90% AVAILABILITY

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3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division I consists of the low-pressure core spray system and low-pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS division II consists of low-pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low-pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI system, 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 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 through a test loop 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. The LPCI system, together with the LPCS system, provides 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 through a test loop 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 III 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 inventory until the vessel is depressurized. The HPCS system operates over a range of 0 to 1177 psi differential pressure between reactor vessel and HPCS suction source.

ECCS - OPERATING and SHUTDOWN (Continued)

The capacity of the HPCS system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 467/1400/5010 gpm at differential pressures of 1177/1147/200 psid. 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 safety 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 will automatically provide makeup, at reactor operating pressures, on a reactor low water level condition. The HPCS out-of-service period of 14 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 through a test loop 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 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety/relief valves although the safety 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 POOL

The supression pool 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 pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

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SUPPRESSION POOL (Continued)

Repair work might require making the suppression pool 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 pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITIONS 4 and 5 the suppression pool minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume, vortex prevention, and a 2' 6" safety margin for conservatism.

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3/4.6.1 CONTAINMENT

3/4.6.1.1 and 3/4.6.1.2 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY (OPERATING and FUEL HANDLING) ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the 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.

3/4.6.1.3 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total primary containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 7.6 psig, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 La, during performance of the periodic tests, to account for possible degradation of the primary 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 to 10 CFR 50.

3/4.6.1.4 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-OPERATING and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.3. 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 primary containment.

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3/4.6.1.5 MAIN STEAM-POSITIVE LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs. Inclusion of the specified leakage control system will prevent untreated leakage from the MSIVs when isolation of the primary system and containment is required. This system includes the Main Steam Shutoff Valves (MSSV).

3/5.6.1.6 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

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 containment will withstand the maximum pressure of 7.6 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment internal pressure ensure that the containment peak pressure of 7.6 psig does not exceed the design pressure of 15.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of + 0.6 psid or the differential at which water would overflow the weir wall into the drywell of 0.58 psid. The limit of -0.3 to + 0.3 psig for initial primary containment internal pressure will limit the peak primary containment pressure to 7.6 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.8 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 185°F during LOCA conditions and is consistent with the safety analysis.

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3/4.6.1.9 PRIMARY CONTAINMENT PURGE SYSTEM

The 36-inch primary containment purge supply and exhaust isolation valves are required to be closed during plant operation, in order to minimize the quantities of radioactive materials released via the containment purge system, except the primary containment purge lines may be opened for up to 2000 hours per 365 days. Additionally, these valves are limited to 65° travel to full open position.

Leakage integrity tests, with a maximum allowable leakage rate for purge supply and exhaust isolation valves, will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage develops. The 0.60 La leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.1.10 PENETRATION VALVE LEAKAGE CONTROL SYSTEM

The OPERABILITY of the penetration valve leakage control system is required to meet the restrictions on overall containment leak rate assumed in the accident analyses.

3/4.6.2 DRYWELL

3/4.6.2.1 DRYWELL INTEGRITY

Drywell integrity ensures that the steam released for the full spectrum of drywell pipe breaks is condensed inside the primary containment either by the suppression pool or by the primary containment ventilation system unit coolers. By utilizing the suppression pool as a heat sink, energy released to the containment is minimized and the severity of the transient is reduced.

3/4.6.2.2 DRYWELL BYPASS LEAKAGE

The limitation on drywell bypass leakage rate is based on having at least one containment ventilation system unit cooler OPERABLE. It ensures that the maximum leakage which could bypass the suppression pool during an accident would not result in the primary containment exceeding its design pressure of 15.0 psig. The integrated drywell leakage value is limited to 10% of the design drywell leakage rate.

The limiting case accident is a very small reactor coolant system break which will not automatically result in a reactor depressurization. The long term differential pressure created between the drywell and primary containment will result in a significant pressure buildup in the primary containment due to this bypass leakage.

3/4.6.2.3 DRYWELL AIR LOCKS

The limitations on closure for the drywell air locks are required to meet the restrictions on DRYWELL INTEGRITY and the drywell leakage rate given in Specifications 3.6.2.1 and 3.6.2.2. 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 the air lock is required to maintain the integrity of the drywell.

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that, during LOCA conditions, the drywell peak pressure of 19.2 psid does not exceed the design pressure of 25.0 psid and the containment peak pressure of 7.6 psig does not exceed the design pressure of 15.0 psig. The maximum external drywell pressure differential is limited to 0.3 psid, well below the 0.58 psid at which suppression pool water will be forced over the wier wall and into the drywell. The limit of 1.2 psid for initial positive drywell-to-containment differential pressure will limit the drywell pressure to 19.2 psid which is less than the design pressure and is consistent with the safety analysis.

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that peak drywell temperature does not exceed the design temperature of 330 °F during LOCA conditions and is consistent with the safety analysis.

3/4.6.2.7 DRYWELL VENT AND PURGE

The 24-in. drywell purge supply and exhaust isolation valves are required to be sealed closed during plant operation, since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. To provide assurance that the 24-inch valves cannot be opened inadvertently, they are sealed closed in accordance with Standard Review Plan Section 6.2.4 by methods that include mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.

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3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and primary containment pressures will not exceed the design pressures of 25 psig and 15 psig, respectively, during primary system blowdown from full operating pressure.

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from 1045 psig. Using conservative parameter inputs, the maximum calculated primary containment pressure during and following a design basis accident is below the primary containment design pressure of 15 psig. Similarly the drywell pressure remains below the design pressure of 25 psig. The maximum and minimum water volumes for the suppression pool are 141,036 cubic feet and 137,571 cubic feet, respectively. These values include the water volume of the primary containment pool, horizontal vents, and weir annulus. Testing in the Mark III Pressure Suppression Test Facility and analysis have assured that the suppression pool temperature will not rise above 185°F for the full range of break sizes.

Should it be necessary to make the suppression pool inoperable, this shall only be done as specified in Specification 3.5.3.

Experimental data indicates that effective steam condensation, without excessive load on the primary containment pool walls, will occur with a quencher device and pool temperature below 200°F during relief valve operation. Specifications have been placed on the envelope of reactor operating conditions to assure the bulk pool temperature does not rise above 185°F in compliance with the containment structural design criteria.

In addition to the limits on temperature of the suppression 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.

The primary containment ventilation system consists of three 100% capacity unit coolers, two of which are safety related. Each of these two unit coolers provides independent 100% heat removal capacity in case of steam bypass of the suppression pool. The turbulence caused by the unit coolers aids in mixing the containment air volume to maintain a homogeneous mixture for H_2 control.

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the

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DEPRESSURIZATION SYSTEMS (Continued)

system is to ensure containment integrity following a LOCA by preventing excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F, considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the standby service water is accomplished in the RHR heat exchangers.

3/4.6.4 PRIMARY CONTAINMENT AND DRYWELL ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the primary containment atmosphere will be isolated from the outside environment, in the event of a release of radioactive material to the primary containment atmosphere or pressurization of the primary containment, and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Primary containment and drywell isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The operability of the drywell isolation valves ensures that the drywell atmosphere will be directed to the suppression pool for the full spectrum of pipe breaks inside the drywell and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Since the allowable value of drywell leakage is so large, individual drywell penetration leakage is not measured. By checking valve operability on any penetration which could contribute a large fraction of the design leakage, the total leakage is maintained at less than the design value.

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 Shield Building Annulus, Auxiliary Building and Fuel Building with associated structures provide secondary containment during normal operation when the containment is sealed and in service.

Establishing and maintaining a vacuum, in the Shield Building Annulus and the Auxiliary Building with the standby gas treatment system and in the Fuel Building with the ventilation system charcoal filtration subsystem, once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate surveillance to ensure the integrity of the secondary containment.

SECONDARY CONTAINMENT (Continued)

The length of time required to reestablish the secondary containment pressure to -0.25 inch W.G. following a LOCA is an important secondary containment functional design element because it is used as an assumption in the radiological analyses. Periodic tests of the secondary containment structure and systems, including the SGTS and the fuel building charcoal filtration system, verify that the measured drawdown times are within specified limits. The drawdown time limits have been established utilizing the same fan performance and building inleakage assumptions as in the post-LOCA analysis except that, since the tests are performed when the plant is shut down, (1) post-LOCA heat loads are not present; (2) the initial secondary containment pressure is atmospheric; and (3) loss of offsite power is not assumed. Meeting these drawdown time limits verifies that secondary containment inleakage and fan performance characteristics will be as assumed in the LOCA analysis. The inleakage values are not verified at these periodic tests since no credit for dilution was taken in the dose calculation.

The OPERABILITY of the standby gas treatment systems and the Fuel Building ventilation charcoal filtration subsystems ensure 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 these systems and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of these systems with the heaters OPERABLE for 10 hours during each 31-day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters.

3/4.6.6 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 containment below its flammable limit during post-LOCA conditions. Either primary containment hydrogen recombiner system with either primary containment/drywell hydrogen mixing 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 primary containment/drywell hydrogen mixing systems are provided to ensure adequate mixing of the primary containment and drywell atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Two 100% capacity primary containment/drywell mixing systems are the primary means of H_2 control within the drywell, exhausting hydrogen produced following a LOCA into the primary containment volume. Hydrogen generated from the metal-water reaction and radiolysis is assumed to evolve to the drywell atmosphere and form a homogeneous mixture through natural forces and from mechanical

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SECONDARY CONTAINMENT (Continued)

turbulence produced by flow from the pipe break. The primary containment/drywell hydrogen mixing system recirculates drywell atmosphere within the primary containment, diluting the drywell hydrogen concentration.

The primary containment/drywell hydrogen mixing valves are required to be closed during plant operation because they have not been demonstrated to be capable of closing during a LOCA or steam-line break accident. However, the 6-inch hydrogen mixing valves may be opened for up to 5 hours per 365 days in OPERATIONAL CONDITIONS 1 and 2 and for up to 90 hours in OPERATIONAL CONDITION 3 for the purpose of controlling drywell pressure. These valves receive an isolation signal in the event of a LOCA and, even if they failed to close, the design drywell bypass leakage of 1.0 sq ft would not be exceeded.

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.

The operability of the primary containment/drywell hydrogen igniters ensures that hydrogen combustion can be accomplished in a controlled manner following a degraded core event that produces hydrogen concentrations in excess of LOCA conditions.

Adjacent igniters are considered to be igniters in different power divisions within approximately 35 feet of each other. Inaccessible areas are defined as areas that have high radiation levels during the entire refueling outage period. These areas are the heat exchanger, filter demineralizer, backwash, and holding pump rooms of the RWCU system.

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3/4.7.1 STANDBY SERVICE WATER SYSTEM

The OPERABILITY of the service water system and ultimate heat sink ensure 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, within acceptable limits, with the assumptions used in the accident analyses.

3/4.7.2 MAIN CONTROL ROOM AIR CONDITIONING SYSTEM

The OPERABILITY of the main control room air conditioning system ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system, in conjunction with control room design provisions, is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3 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 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3, when reactor vessel pressure exceeds 150 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 which justifies the specified 14 day out-of-service period.

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REACTOR CORE ISOLATION COOLING SYSTEM (Continued)

The surveillance requirements provide adequate assurance that RCIC 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.

3/4.7.4 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event that initiates dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, 2-kip, 10-kip, and 100-kip capacity mechanical snubbers utilizing the same design features and manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" would be of a different type, for the purposes of this Technical Specification, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location, size and system affected shall be available at the plant. The accessibility of each snubber shall be determined and approved by the Facility Review Committee. The determination shall be based upon the accessibility of the snubber during plant operations (e.g., radiation level, temperature, atmosphere, location, etc.) The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

When a snubber is found inoperable, an engineering evaluation is performed, including the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

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. Selection of a representative sample according to one of the four sample plans provides assurance that the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers will require functional testing of additional units.

The service life of a snubber is evaluated from manufacturer input and from consideration of the snubber service conditions and associated installation and maintenance records ,i.e., newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . . . The requirement to monitor the snubber service life is included to ensure

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SNUBBERS (Continued)

that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and of the snubber service life review are not intended to affect plant operation.

3/4.7.5 SEALED SOURCE CONTAMINATION

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

3/4 7.6 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 systems consist of the water system, spray and/or sprinkler systems, Halon system 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 assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying the weight and pressure of the tanks.

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.

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3/4.7.7 FIRE-RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 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.

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis in FSAR Chapter 15.

3/4 7.10 STRUCTURAL SETTLEMENT

Structural settlement limitations are imposed and required to be verified so as to preserve the assumptions made in the static design of the major safety related structures.

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3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

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

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources is consistent with the initial condition assumptions of the safety analyses and is based upon maintaining OPERABLE at least Division I or II of the onsite A.C. and D.C. power sources and associated distribution systems during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. or D.C. source. Division III supplies the high pressure core spray (HPCS) system and the standby service water pump 1SWP*P2C with its auxiliaries.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When diesel generator 1A or 1B is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator 1A or 1B as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power, during the period that diesel generator 1A or 1B is inoperable, will not result in a complete loss of safety function of critical systems. The term verify, as used in this context, means to administratively check, by examining logs or other information, to determine if certain components are out of service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

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

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory 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.

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A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirements 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, and IEEE Std 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

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

Table 4.8.2.1-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's nominal full charge specific gravity, or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates so that they retain an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations, penetration conductors, main control room lighting, and RPS alternate source of power are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of the overcurrent protection circuit breakers and/or motor starters by periodic surveillance.

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

Specific surveillance tests on the bypass circuits of motor operated valves' thermal overload protection is not required at River Bend because the circuits are integral with the starting circuits of the motor operated valves and are therefore tested during functional tests of the valves. For the motor operated valve thermal overloads not bypassed, the thermal overloads are tested under Specification 4.8.4.2.a.3. These surveillance requirements are in accordance with RG 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

The reactor protection system (RPS) electric power monitoring assemblies provide redundant protection to the RPS, and to other systems that receive power from the RPS buses, by acting to disconnect the RPS from the power source circuits in the presence of an electrical fault in the power supply.

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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 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 safety analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling platform 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.

3/4.9.6 REFUELING PLATFORM

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 hoist has sufficient load capacity for handling fuel assemblies and/or 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.

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3/4.9.7 CRANE TRAVEL - SPENT AND NEW FUEL STORAGE, TRANSFER AND UPPER CONTAINMENT FUEL POOLS

The restriction on movement, over other fuel assemblies in the pools, of loads in excess of the nominal weight of a fuel assembly, ensures that, in the event this load is dropped, the activity release will be limited to that contained in 123 fuel rods and any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

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 safety 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 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE and in operation or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and (2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

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3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

The purpose of the inclined fuel transfer system specification is to control personnel access to those potentially high radiation areas immediately adjacent to the system and to assure safe operation of the system.

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3/4.10.1 PRIMARY CONTAINMENT INTEGRITY/DRYWELL INTEGRITY

The requirements for PRIMARY CONTAINMENT INTEGRITY and DRYWELL INTEGRITY are not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD PATTERN 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 requirements 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 permit performance of certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

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 low temperature, while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode. This arrangement minimizes contaminated water discharge to the radioactive waste disposal system.

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3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope, and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually); Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968); and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials

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3/4.11.1.2 DOSE (Continued)

in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and in Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The requirement that the appropriate portions of this system be used when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this Specification include those unprotected outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that, in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose, at any time at and beyond the SITE BOUNDARY, from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC

3/4.11.2.1 DOSE RATE (Continued)

will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous effluent samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually); Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem 40, 586-93 (1968); and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

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3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 AND 3/4.11.2.5 GASEOUS RADWASTE TREATMENT AND VENTILATION EXHAUST TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

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3/4 11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the main condenser offgas treatment system is maintained below the flammability limits of hydrogen. Automatic control features are included in the system to prevent the hydrogen concentration from reaching this flammability limit. These automatic control features include isolation of the source of hydrogen or injection of dilutants to reduce the concentration below the flammability limit. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criteria 3 and 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MAIN CONDENSER

Restricting the release rate of the sum of the activities from the noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

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3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant-generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report, with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected) in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance relates only to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials, resulting from the station operation, in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially-specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12.1-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually); Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968); and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Handford Company Report ARH-SA-215 (June 1975).

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3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities, shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) the vegetation yield is 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

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SECTION 5.0

DESIGN FEATURES

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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.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1.1-1 and 5.1.3-1.

5.2 CONTAINMENT

PRIMARY CONTAINMENT

5.2.1 The primary containment is a steel structure composed of a vertical right cylinder and a torispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head. Primary containment contains approximately 20 feet of water in the suppression pool which is connected to the drywell through a series of horizontal vents. The primary containment has a minimum net free air volume of 1,190,000 cubic feet. The drywell has a minimum net free air volume of 236,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure:
   1. Drywell 25 psig.
   2. Primary Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

c. Maximum external-to-internal differential pressure:
   1. Drywell 20 psid.
   2. Primary Containment 0.6 psid.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the shield building, the auxiliary building and the fuel building. Secondary containment has a minimum free volume of 2,249,400 cubic feet.

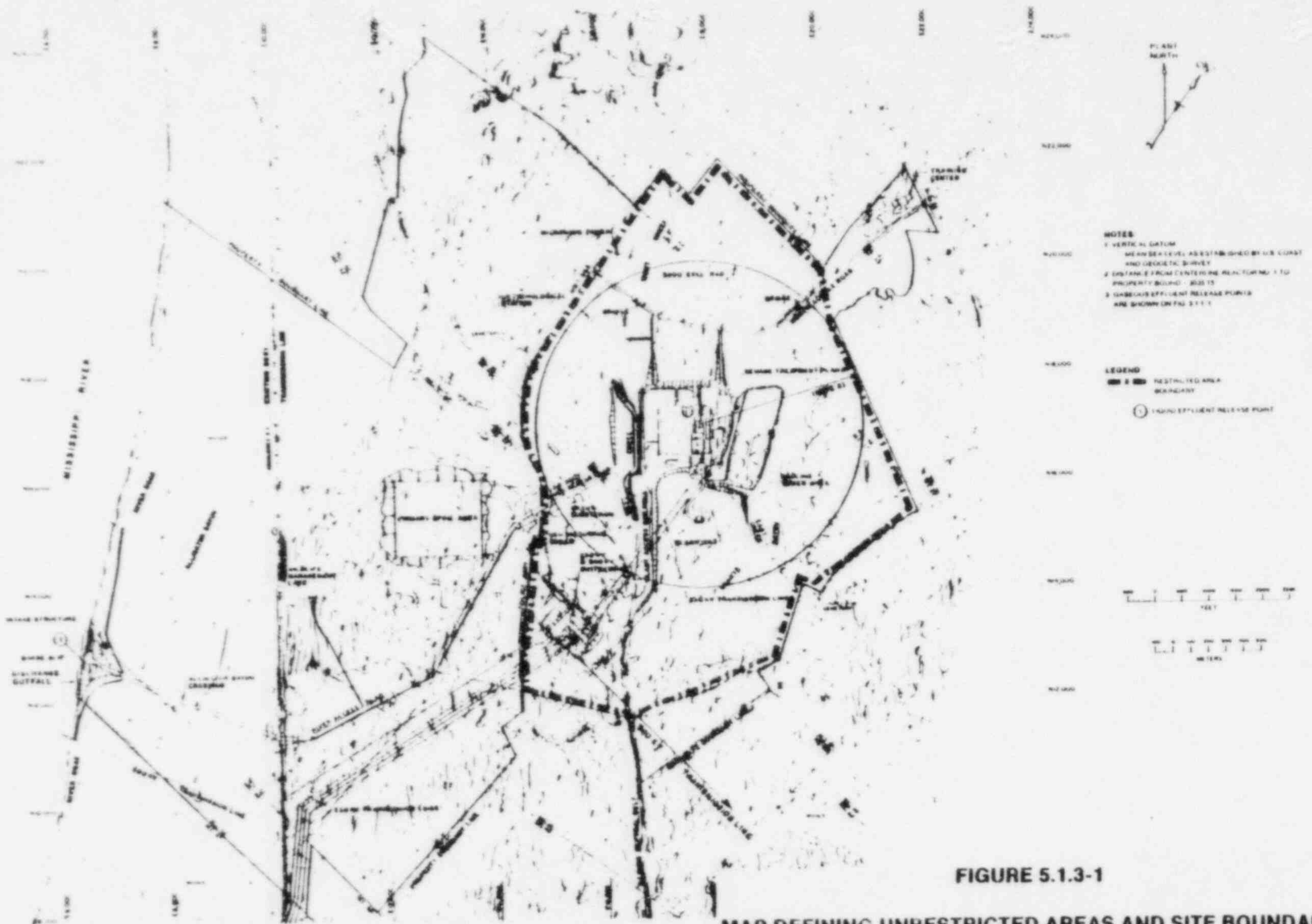

FIGURE 5.1.3-1

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

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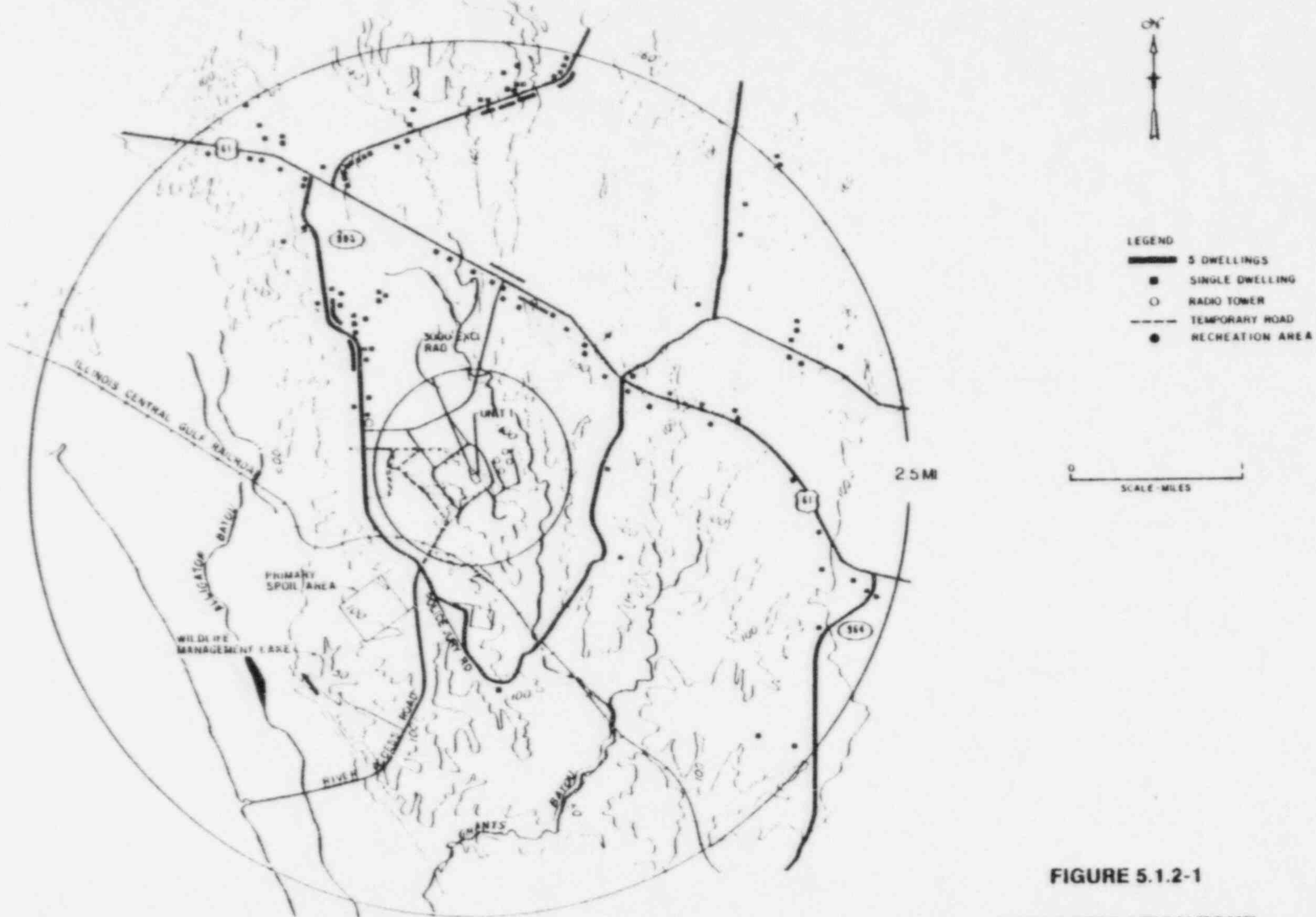

FIGURE 5.1.2-1

LOW POPULATION ZONE

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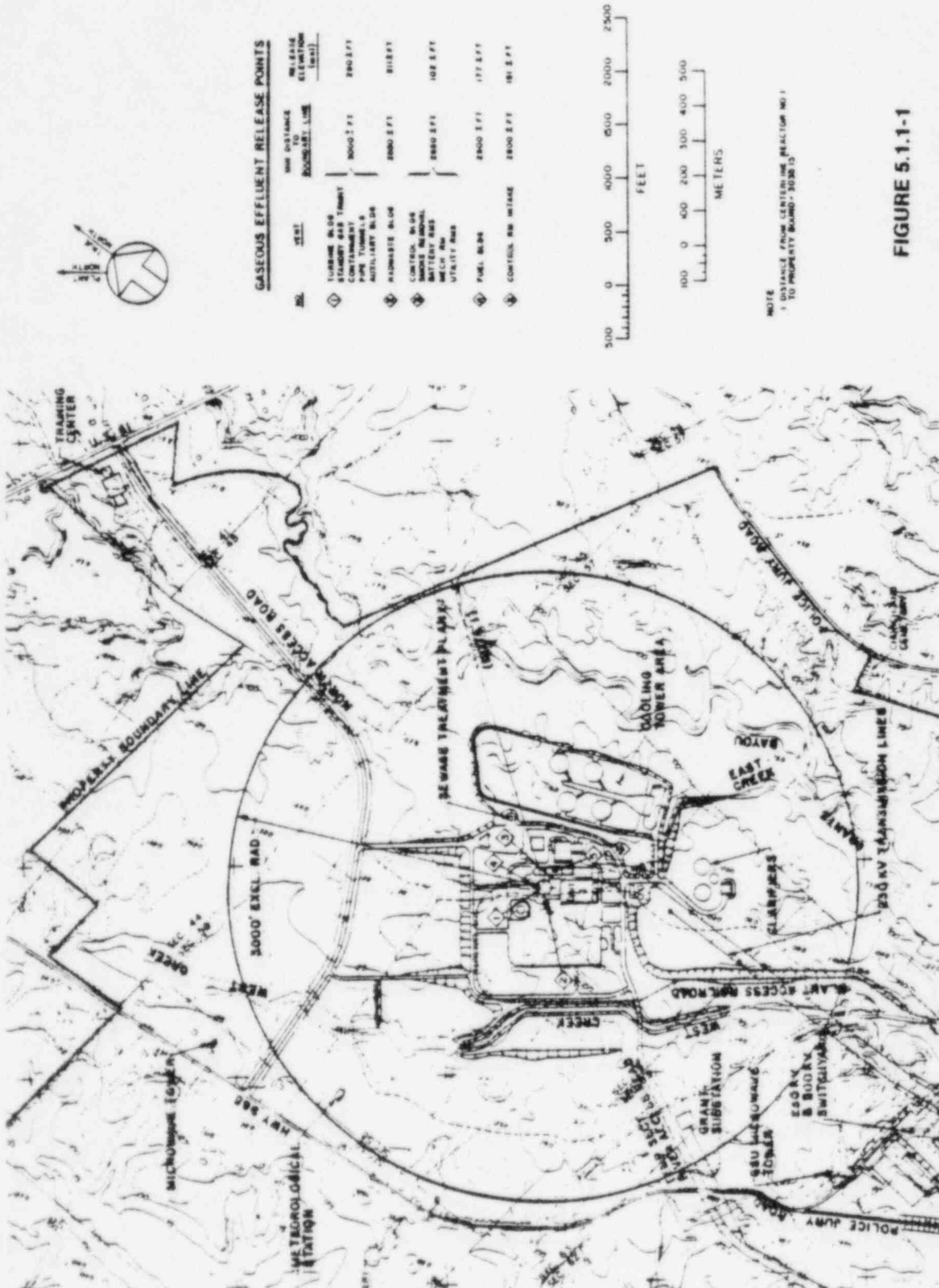

FIGURE 5.1.1-1

EXCLUSION AREA

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 624 fuel assemblies. Each fuel assembly shall contain 62 fuel rods and two water rods, all clad with Zircaloy-2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.70 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 145 control rod assemblies, each consisting of a cruciform array of stainless steel tubes surrounded by a cruciform shaped stainless steel sheath. Each tube shall contain 143.7 inches of boron carbide (B_4C) powder.

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:

   1. 1250 psig on the suction side of the recirculation pump.

   2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   3. 1550 psig from the discharge shutoff valve to the jet pumps.

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.

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

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

a. A k_{eff} less than or equal to 0.95, when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1 of the FSAR.

b. A fuel assembly center-to-center storage spacing of 7 in. within rows and 12.25 in. between rows in the Low Density Storage Racks.

c. A fuel assembly center-to-center storage spacing of 6.28 in., with a neutron poison material between storage spaces, in the High Density Storage Racks.

The storage of spent fuel in the upper containment fuel storage pool is prohibited during OPERATIONAL CONDITIONS 1 and 2.

5.6.1.2 For the first core loading, the K_{eff} for new fuel stored dry in the spent fuel storage racks shall be administratively controlled to not exceed 0.98 when optimum moderation (foam, spray, fogging, or small droplets) is assumed.

5.6.1.3 Provisions shall be taken to avoid the entry of sources of optimum moderation (foam, spray, fogging, or small droplets) to preclude that K_{eff} for new fuel, stored in the new fuel storage facility, could exceed 0.98.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 95'.

CAPACITY

5.6.3 The spent fuel storage pool in the fuel building is designed and shall be maintained with a storage capacity limited to no more than 2680 fuel assemblies. Only fuel manufactured by General Electric may be stored in the spent fuel pool.

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.

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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
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	40 hydrostatic pressure or leak tests	Pressurized to $\geq$ 930 psig and $\leq$ 1250 psig

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Senior Vice President - River Bend Nuclear Group, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown 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. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room;

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least five members shall be maintained on site at all times*. The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, the Control Operating Foreman, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Radiation Protection Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 42-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or either one of the Assistant Plant Managers or the Supervisor-Radiological Programs, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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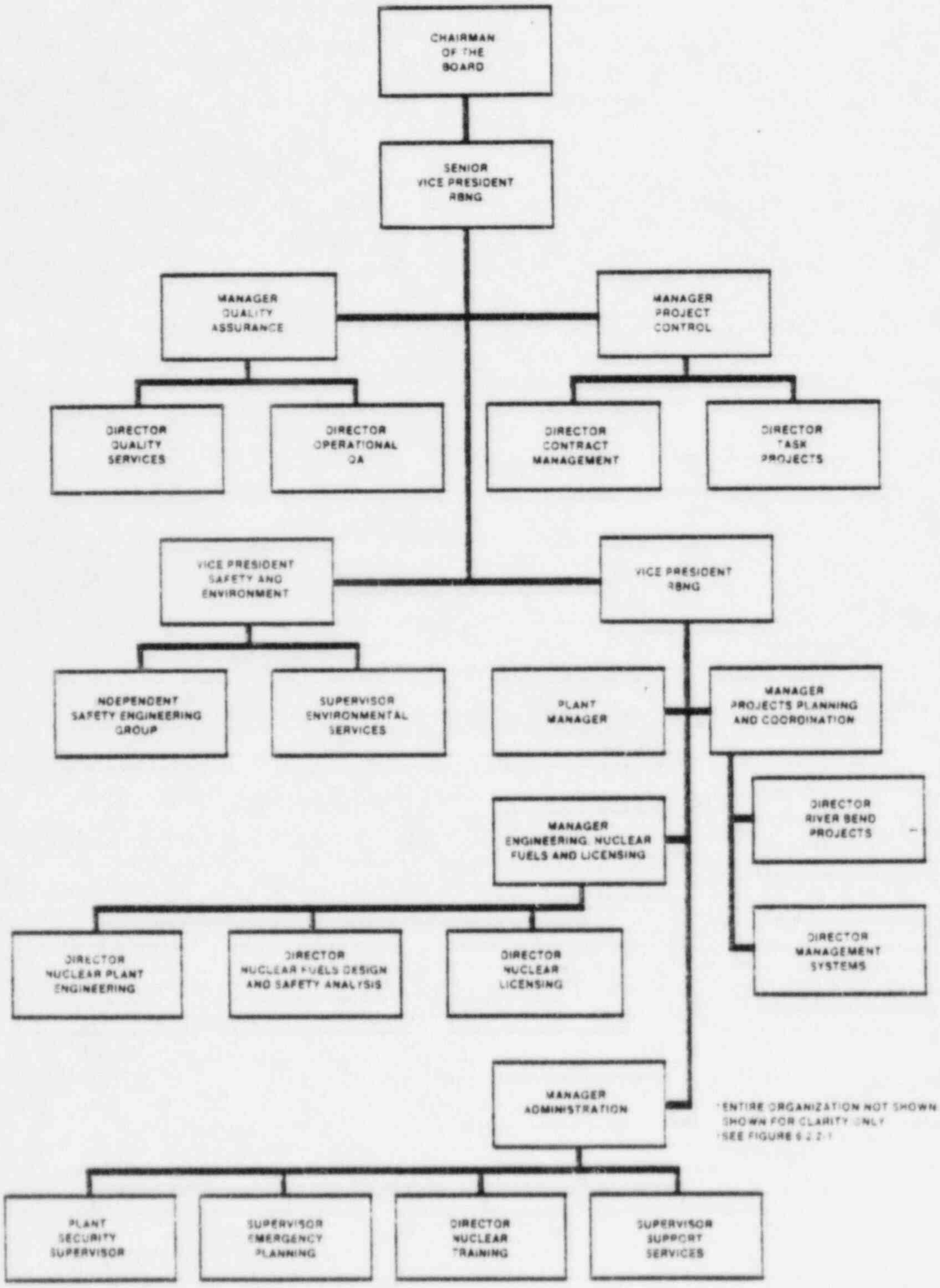

FIGURE 6.2.1-1
RBNG ORGANIZATION

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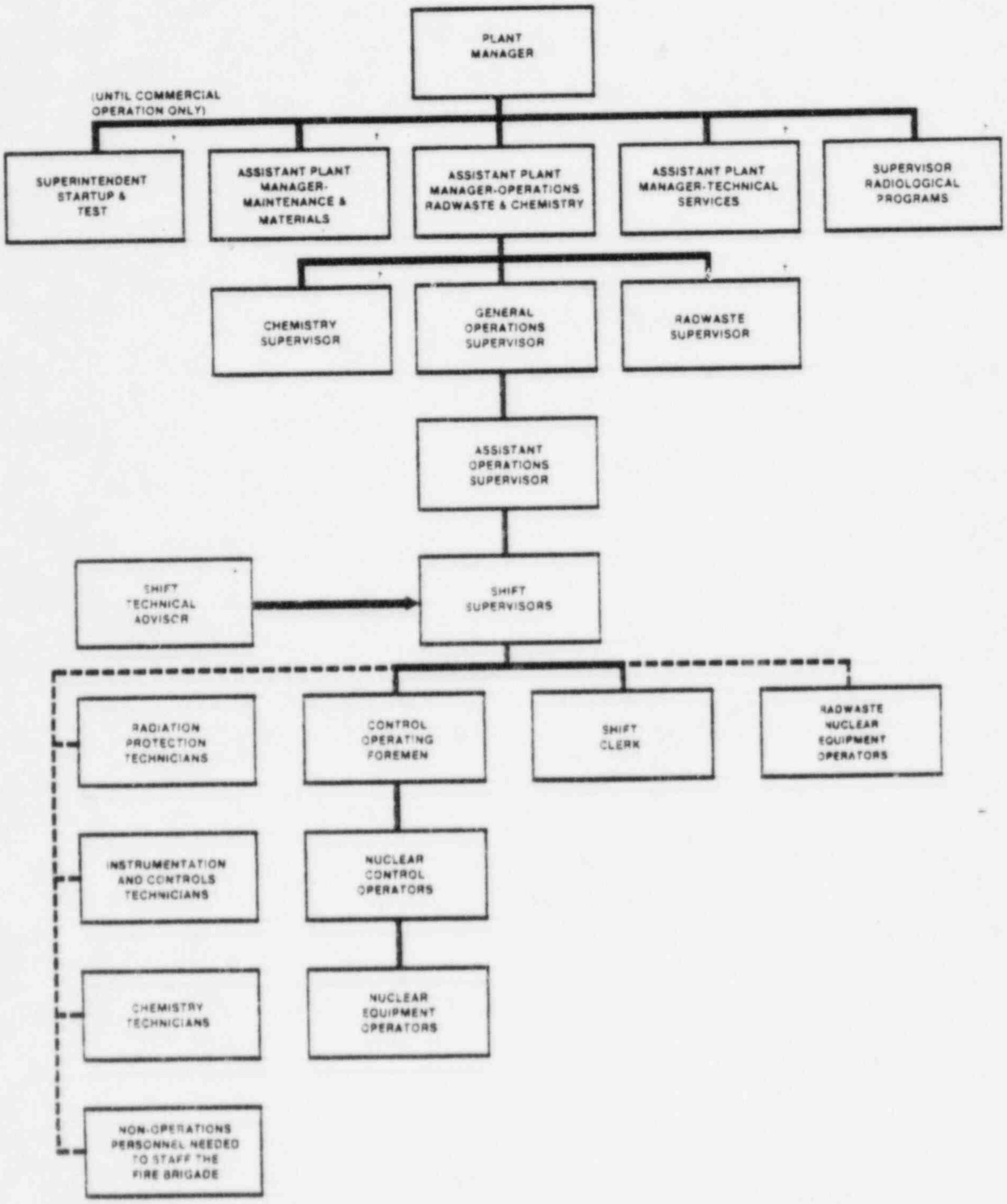

- - - - THESE INDIVIDUALS ARE ON-SHIFT BUT DO NOT REPORT DIRECTLY TO THE SHIFT SUPERVISOR UNLESS HE REPRESENTS THE SENIOR MANAGEMENT ON SITE.
*THE SHIFT TECHNICAL ADVISOR (STA) POSITION MAY BE FILLED BY AN ON-SHIFT SHIFT SUPERVISOR OR CONTROL OPERATING FOREMAN PROVIDED THE INDIVIDUAL MEETS THE STA QUALIFICATIONS OF SPECIFICATION 6.2.4 AND 5 LICENSED INDIVIDUALS ARE ON-SHIFT.
†ORGANIZATION NOT SHOWN.

FIGURE 6.2.2-1
RIVER BEND STATION ORGANIZATION

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TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA*	1	None

TABLE NOTATION

SS - Shift Supervisor with a Senior Operator license on Unit 1.
SRO - Individual with a Senior Operator license on Unit 1.
RO - Individual with an Operator license on Unit 1.
AO - Auxiliary operator
STA - Shift Technical Advisor

*The Shift Technical Advisor (STA) position may be filled by an on-shift Shift Supervisor (SS) or Senior Reactor Operator (SRO) provided the individual meets the STA qualifications of Specification 6.2.4 and five (5) licensed operators are on shift.

The shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Vice President - Safety and Environment.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located onsite. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, of which at least 1 year experience shall be in the nuclear field.

RESPONSIBILITIES

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

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Vice President - Safety and Environment.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

For the dual role position shown in Table 6.2.2-1, the Shift Technical Advisor shall have a bachelor's degree or shall have completed all technical courses required for the degree in a scientific or engineering discipline and shall have received all of the training for the normal STA position described above.

*Not responsible for sign-off function.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the Supervisor-Radiological Programs who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director - Nuclear Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 FACILITY REVIEW COMMITTEE (FRC)

FUNCTION

6.5.1.1 The FRC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The FRC shall be composed of the:

Chairman:	Assistant Plant Manager-Technical Services
Member:	Assistant Plant Manager-Operations, Radwaste and Chemistry
Member:	Assistant Plant Manager-Maintenance and Material
Member:	General Operations Supervisor
Member:	Supervisor-Radiological Program
Member:	Reactor Engineering Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the FRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in FRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The FRC shall meet at least once per calendar month and as convened by the FRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the FRC necessary for the performance of the FRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including no more than two alternates.

RESPONSIBILITIES

6.5.1.6 The FRC shall be responsible for:

a. Review of all plant general administrative procedures and changes thereto;

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 structures, components, systems or equipment that affect nuclear safety;

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 Vice President - RBNG and the Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety; items that may be included in this review are NRC inspection reports that require written response, QA audits/surveillance findings of operating and maintenance activities, NRB audit results, and American Nuclear Insurer (ANI) inspection results;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager or the Nuclear Review Board;

i. Review of initial start-up testing phase start-up procedures and revisions; and

j. Review of the Emergency Plan and implementation procedures at least once per 12 months and all proposed changes thereto.

6.5.1.7 The FRC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6.a. through d. prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6.a. through e. constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President - RBNG and the Nuclear Review Board of disagreement between the FRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

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RECORDS

6.5.1.8 The FRC shall maintain written minutes of each FRC meeting that, at a minimum, document the results of all FRC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Plant Manager and the NRB.

6.5.2 TECHNICAL REVIEW AND CONTROL

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group that prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group that prepared the procedure. Each such procedure and program, or changes thereto, shall be approved, prior to implementation, by the Plant Manager or one of the Assistant Plant Managers or the Supervisor - Radiological Programs, with the exception of the Emergency Plan and implementing procedures which shall be approved by the Manager - Administration, Plant Manager and Vice President - RBNG.

6.5.2.2 Individuals responsible for reviews performed in accordance with Section 6.5.2.1 shall be members of River Bend Nuclear Group supervisory staff, and the reviews shall be performed in accordance with administrative procedures. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary and a verification that the proposed actions do not constitute an unreviewed safety question. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.3 The station security program and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Specification 6.5.2.1.

6.5.2.4 The station emergency plan and implementing procedures and recommended changes shall be approved in accordance with Specification 6.5.2.1.

6.5.2.5 The station fire protection plan and implementing procedures shall be reviewed at least once per 12 months, and recommended changes approved in accordance with Specification 6.5.2.1.

6.5.2.6 Records documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.5 shall be maintained.

6.5.3 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.3.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

FUNCTION (Continued)

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. Licensing and regulatory affairs,
j. Training.

The NRB shall report to and advise the Senior Vice President - RBNG on those areas of responsibility in Specifications 6.5.3.7 and 6.5.3.8.

COMPOSITION

6.5.3.2 The NRB shall be composed of the:

Chairman:	Vice President - Safety and Environment
Member and Vice Chairman	Vice President - RBNG
Member:	Manager - Administration
Member:	Executive Vice President-External Affairs and Production
Member:	Manager-Design Engineering, Technical Services Department
Member:	Manager-Engineering, Nuclear Fuels, and Licensing
Member:	Plant Manager
Member:	Assistant Plant Manager-Operations, Radwaste and Chemistry
Member:	Manager-Quality Assurance
Member:	Director-Nuclear Licensing
Member:	Director-Nuclear Plant Engineering
Member:	Director-Nuclear Fuels Design and Safety Analysis

ALTERNATES

6.5.3.3 All alternate members shall be appointed in writing by the NRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NRB activities at any one time.

CONSULTANTS

6.5.3.4 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

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MEETING FREQUENCY

6.5.3.5 The NRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.3.6 The quorum of the NRB necessary for the performance of the NRB review and audit functions of these Technical Specifications shall consist of the Chairman or the Vice Chairman and at least six NRB members including no more than two alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.3.7 The NRB shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the FRC.

AUDITS

6.5.3.8 Audits of unit activities shall be performed under the cognizance of the NRB. These audits shall encompass:

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AUDITS (Continued)

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions, at least once per 12 months;

b. The performance, training and qualifications of the entire unit staff, at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures, at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation, at least once per 12 months, utilizing either qualified offsite licensee personnel or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least every third year; and

g. Any other area of unit operation considered appropriate by the NRB, Plant Manager or the Senior Vice President - RBNG.

h. The Emergency Plan and implementing procedures, at least once per 12 months.

i. The Security Plan and implementing procedures, at least once per 12 months.

j. The radiological environmental monitoring program and the results thereof, at least once per 12 months.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

l. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes, at least once per 24 months.

m. The performance of activities by the Quality Assurance Program for effluent and environmental monitoring, at least once per 12 months.

RECORDS

6.5.3.9 Records of NRB activities shall be prepared, approved, and distributed as indicated below:

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RECORDS (Continued)

a. Minutes of each NRB meeting shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.3.7 shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.3.8 shall be forwarded to the Senior Vice President - RBNG, and to the management positions responsible for the areas audited, within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73 and

b. Each REPORTABLE EVENT shall be reviewed by the FRC and the results of this review shall be submitted to the NRB and the Plant Manager.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President - RBNG and the NRB chairman (or personnel acting for their function) shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRC. 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.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NRB, and the Senior Vice President - RBNG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

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PROCEDURES AND PROGRAMS (Continued)

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. Process Control Program implementation.

i. Offsite Dose Calculation Manual implementation.

j. Quality Assurance Program for effluent and environmental monitoring.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed and approved in accordance with Specification 6.5.2.1.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed by the FRC as required by Specification 6.5.1.6, and approved in accordance with Specification 6.5.2.1 within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage, from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident, to as low as practicable levels. The systems include the HPCS, LPCS, RHR, RCIC, process sampling, and standby gas treatment systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

PROCEDURES AND PROGRAMS (Continued)

2. Integrated leak test requirements for each system, at refueling cycle intervals or more frequently.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

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

c. Post-accident Sampling

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

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

d. Biofouling Prevention and Detection

A program, approved by the NRC Staff prior to introduction of river water to the systems, which will include the procedures to prevent biofouling of safety-related equipment, to assure detection of Corbicula in the intake embayment and the Mississippi River at the River Bend Station site, and to monitor and survey safety-related equipment to detect biofouling. Changes to this program will be submitted to and approved by the NRC (both the Region and NRR) prior to implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to

STARTUP REPORT (Continued)

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.

6.9.1.2 The startup report shall address each of the tests identified in Final Safety Analysis Report (Section 14.2.12.2) and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. Documentation of all challenges to safety/relief valves and a summary of SRV failures.

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all failures to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate) with preoperational studies, operational controls and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of all analyses of radiological environmental samples and of all environmental radiation measurements taken during the report period pursuant to the locations specified in the tables and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 4.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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SEMIANNUAL EFFLUENT RELEASE REPORT

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appedix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction and atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.3) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters of the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most-exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

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*In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

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SEMIANNUAL EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the ODCM, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2

SPECIAL REPORTS

6.9.2 Special reports shall be submitted in the following manner:

a. Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

b. Special reports in regard to Corbicula will be submitted to the NRC within 30 days of identification of infestation. In accordance with the settlement agreement dated October 10, 1984, these reports shall describe the level of infestation, affected systems and measures taken to prevent recurrence.

6.10 RECORD RETENTION

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

6.10.2 The following records shall be retained for at least 5 years:

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

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

c. All REPORTABLE EVENTS

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

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

f. Records of radioactive shipments.

RECORD RETENTION (Continued)

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.

i. Records of emergency drills and exercises.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

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

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

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

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

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

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

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Operational Quality Assurance Manual that are not listed in Specification 6.10.1.

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 FRC and the NRB.

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

RECORD RETENTION (Continued)

m. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at the specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/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 (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

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 levels in the area have been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and who shall perform periodic radiation surveillance at the frequency specified in the RWP by the Health Physicist.

6.12.2 In addition to the requirements of Specification 6.12.1, accessible areas with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Control Operating Foreman on duty and/or the radiation protection supervision. Doors shall remain locked except during periods of

*Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

access under an approved RWP that specifies the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For accessible areas that are located within large areas, such as the containment, where no enclosure exists for purposes of locking and no enclosure can be reasonably constructed around the individual areas, and within which radiation levels are such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrem,* then that area shall be roped off and conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

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

6.13.2 Licensee-initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to Specification 6.5.2.

2. Shall become effective upon review and acceptance pursuant to Specification 6.5.2.

*Measurement made at $\leq$ 18 inches from source of radioactivity.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee-initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed, with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable pursuant to Specification 6.5.2.

2. Shall become effective upon review and acceptance pursuant to Specification 6.5.2.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation of the change was reviewed and approved pursuant to Specification 6.5.2. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59.

   b. Sufficiently detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows (1) the predicted releases of radioactive materials in liquid and gaseous effluents and/or (2) quantity of solid waste, that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures, to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population, that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable pursuant to Specification 6.5.2.

2. Shall become effective upon review and acceptance pursuant to Specification 6.5.2.