PDR

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

July 27, 1984

MEMORANDUM FOR: Those on Attached List

FROM: Dennis Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

SUBJECT: PROOF AND REVIEW OF ENRICO FERMI - UNIT 2
TECHNICAL SPECIFICATIONS

Attached please find the Appendix A Technical Specifications for Enrico Fermi - Unit 2 for proof and review. We request that you review those sections of the attached Technical Specifications which pertain to your particular areas of responsibility and that the results of this review, identifying those sections of the Technical Specifications reviewed, be forwarded to me by August 31, 1984.

Mr. Donald R. Hoffman, of my staff, will be available during this review period to answer any questions your staff may have. He is located in Room 287, Phillips, and his telephone extension is 49-28518.

If you have comments or suggestions, or if you are in agreement with the Technical Specifications content in your area of review, it is requested that a written response to that effect identifying those sections of the Technical Specifications reviewed be provided by the above specified date.

By copy of this memorandum, the Assistant Director for Licensing, DL is requested to have the Chief for LB#1, DL forward three copies of the enclosed technical specifications to the applicant for review.

Dennis M Crutchfield
Dennis Crutchfield, Assistant Director
for Safety Assessment
Division of Licensing

Enclosure:
Enrico Fermi - Unit 2
Technical Specifications

cc: w/o enclosure
D. Eisenhut

w/ enclosure
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8408060388 XA

Proof & Review Distribution
for Enrico Fermi Unit 2
Date: July 27, 1984

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DETROIT EDISON COMPANY

ENRICO FERMI ATOMIC POWER PLANT

UNIT 2

(FERMI - UNIT 2)

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

1.1 ACTION shall be 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 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.

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

1.12 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

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

FREQUENCY NOTATION

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

IDENTIFIED LEAKAGE

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

ISOLATION SYSTEM RESPONSE TIME

1.16 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

1.17 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

1.18 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

1.19 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, 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.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

MEMBER(S) OF THE PUBLIC

1.21 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

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

OFF-GAS TREATMENT SYSTEM

1.23 An OFF-GAS TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting reactor coolant system offgases from the reactor coolant and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OFFSITE DOSE CALCULATION MANUAL

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

OPERABLE - OPERABILITY

1.25 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

1.26 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

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

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

e. The suppression chamber is in compliance with the requirement of Specification 3.6.2.1.

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

g. The suppression chamber to reactor building vacuum breakers are in compliance with Specification 3.6.4.2

THE PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that 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 Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

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

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3292 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.33 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

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

ROD DENSITY

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

SECONDARY CONTAINMENT INTEGRITY

1.36 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

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

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

SHUTDOWN MARGIN

1.37 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

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.41 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

1.43 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.45 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.

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VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

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.
NA	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours 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 785 psig and core flow greater than 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 785 psig and core flow greater than 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, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours 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.

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-Upscale, Setdown	$\leq$ 15% of RATED THERMAL POWER	$\leq$ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	$\leq$ 0.66 W+51%, with a maximum of	$\leq$ 0.66 W+54%, with a maximum of
2) High Flow Clamped	$\leq$ 113.5% of RATED THERMAL POWER	$\leq$ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	$\leq$ 118% of RATED THERMAL POWER	$\leq$ 120% of RATED
d. Inoperative	N.A.	N.A.
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 1068 psig	$\leq$ 1088 psig
4. Reactor Vessel Low Water Level - Level 3	$\geq$ 173.4 inches*	$\geq$ 171.9 inches
5. Main Steam Line Isolation Valve - Closure	$\leq$ 8% closed	$\leq$ 12% closed
6. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
7. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	$\leq$ 120 gallons**	$\leq$ 160 gallons**
b. Level Transmitter	$\leq$ 100 gallons**	$\leq$ 160 gallons**
9. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
10. Turbine Control Valve Fast Closure	Initiation of fast closure	N.A.
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.
13. Backup Manual Scram	N.A.	N.A.

*See Bases Figure B 3/4 3-1.
**Volume is from closed drain valve C11-F011.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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.

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 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

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

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

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

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.

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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.

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

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 1968 Edition, including Addenda through Summer 1969, 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 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 shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety Limit has been established at the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action.

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2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

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

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15B.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

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

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale 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 Neutron Flux-High setpoint, a time constant of 6 ± 1 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 MFLPD is greater than or equal 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 stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

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REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Low Water Level-Level 3

The reactor vessel water level trip setpoint was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Main Steam Line Isolation Valve-Closure

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

6. Main Steam Line Radiation-High

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

7. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a complete 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. The trip setting was selected as low as possible without causing spurious trips.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for the combined scram discharge volume is equivalent to a contained volume of 120 gallons of water.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 6% 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.

10. Turbine Control Valve Fast Closure

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 turbine control valve (TCV) fast closure signal is generated independently in each valve control logic and connected directly to the Reactor Protection System. The signal to the Reactor Protection System is generated simultaneously with the deenergizing of the solenoid dump valves which produces control valve fast closure. Therefore, when TCV fast closure occurs, a scram trip signal is initiated.

11. Reactor Mode Switch Shutdown Position

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

12. Manual Scram

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

13. Backup Manual Scram

The Backup Manual Scram is a diverse method for manual scram and provides a second means for manual reactor trip capability.

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL 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, within one hour action shall be initiated 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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APPLICABILITY

SURVEILLANCE REQUIREMENTS

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:

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

With the SHUTDOWN MARGIN less than specified:

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

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

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

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 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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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 1 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 1 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*.

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.

---

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

**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 (Continued)

ACTION: (Continued)

2. If the inoperable control rod(s) is inserted, within 1 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.

d. With one scram discharge volume vent valve and/or one scram discharge volume drain valve inoperable and open, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

e. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore the inoperable valve(s) to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 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 preset power level of the RWM and RSCS, 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.

---

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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 float response by verification of proper float switch actuation after each scram from a pressurized condition greater than or equal to 900 psig.

4.1.3.1.5 All control rod drives, when removed for maintenance or for access to the in-service CRD housing inspections, shall have a dye penetrant examination of the collet retainer tube outer surface.

*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided the surveillance is performed within 12 hours after achieving less than or equal to 50% ROD DENSITY.

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 6, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

   1. Declare the control rod(s) with the slow insertion time inoperable, and

   2. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds.

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

b. The provisions of Specification 3.0.4 are not applicable.

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.

d. The provisions of Specification 4.0.4 are not applicable.

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CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

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

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
46	0.358
36	1.096
26	1.860
6	3.419

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

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

Position Inserted From Fully Withdrawn	Average Scram Insertion Time (Seconds)
46	0.379
36	1.161
26	1.971
6	3.624

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

1. Declare the control rods with the slower than average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

2. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit.

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

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

ACTION:

a. In OPERATIONAL CONDITION 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 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 1 hour, either:

      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 or 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.

---

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 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 940 + 30, -0 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.

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

   1. If permitted by the RWM and RSCS, 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 RWM or RSCS, then until permitted by the RWM and RSCS, 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

   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.

---

*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.

SURVEILLANCE REQUIREMENTS

4.1.3.6 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.

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

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

a) Determine the position of the control rod by:

1) Moving the control rod, by single notch movement, to a position with an OPERABLE position indicator,

2) Returning the control rod, by single notch movement, to its original position, and

3) Verifying no control rod drift alarm at least once per 12 hours, or

b) Move the control rod to a position with an OPERABLE position indicator, or

c) When THERMAL POWER is:

1) Within the preset power level of the RSCS, declare the control rod inoperable, or

2) Greater than the preset power level of the RSCS, 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.

*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.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

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

      1) Within 1 hour:

         (a) Determine the position of the control rod(s) per ACTION a.1.a), above, or

         (b) Move the control rod to a position with an OPERABLE position indicator, or

         (c) Declare the control rod inoperable.

      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 member of the unit technical staff.

   b) When THERMAL POWER is greater than the preset power level of the RSCS, determine the position of the control rod(s) per ACTION a.1.a), above.

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

b. In OPERATIONAL CONDITION 5* with a withdrawn control rod position indicator inoperable, move the control rod to a position with an OPERABLE position indicator or insert the control rod.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*, when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER, the minimum allowable preset power level.

ACTION:

a. With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

b. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

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

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

---

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

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to 20% RATED THERMAL POWER, the minimum allowable preset power level.

ACTION:

a. With the RSCS inoperable, control rod movement shall not be permitted, except by a scram.

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

   1. The position and bypassing of inoperable control rods is verified by a second licensed operator or other technically qualified member of the unit technical staff, and

   2. There are not more than 3 inoperable control rods in any RSCS group.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

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

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

   2. Within 1 hour after rod inhibit mode automatic initiation when reducing THERMAL POWER.

b. Attempting to move a control rod more than one notch prior to other control rod movement after the group notch mode is automatically initiated during:

   1. Control rod withdrawal for each reactor startup, and

   2. Power reduction.

c. Performance of the comparator check of the group notch circuits within 8 hours prior to control rod:

   1. Withdrawal for each reactor startup, and

   2. Insertion to reduce THERMAL POWER to less than 20% of RATED THERMAL POWER.

---

*See Special Test Exception 3.10.2

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

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

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

ACTION:

a. With one RBM channel inoperable:

    1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

    2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

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

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

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

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3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 5*

ACTION:

a. In OPERATIONAL CONDITION 1 or 2:

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

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

b. In OPERATIONAL CONDITION 5*:

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

2. With the standby liquid control system otherwise inoperable, insert all insertable control rods within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

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

1. The temperature of the sodium pentaborate solution is greater than or equal to 70°F.

2. The available volume of sodium pentaborate solution is within the limits of Figure 3.1.5-1.

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

---

*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)

b. At least once per 31 days by:

   1. Verifying the continuity of the explosive charge.

   2. Determining that the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

   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 at a pressure of greater than or equal to 1190 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 charge of that batch successfully fired. Both injection loops shall be tested in 36 months.

   2. Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

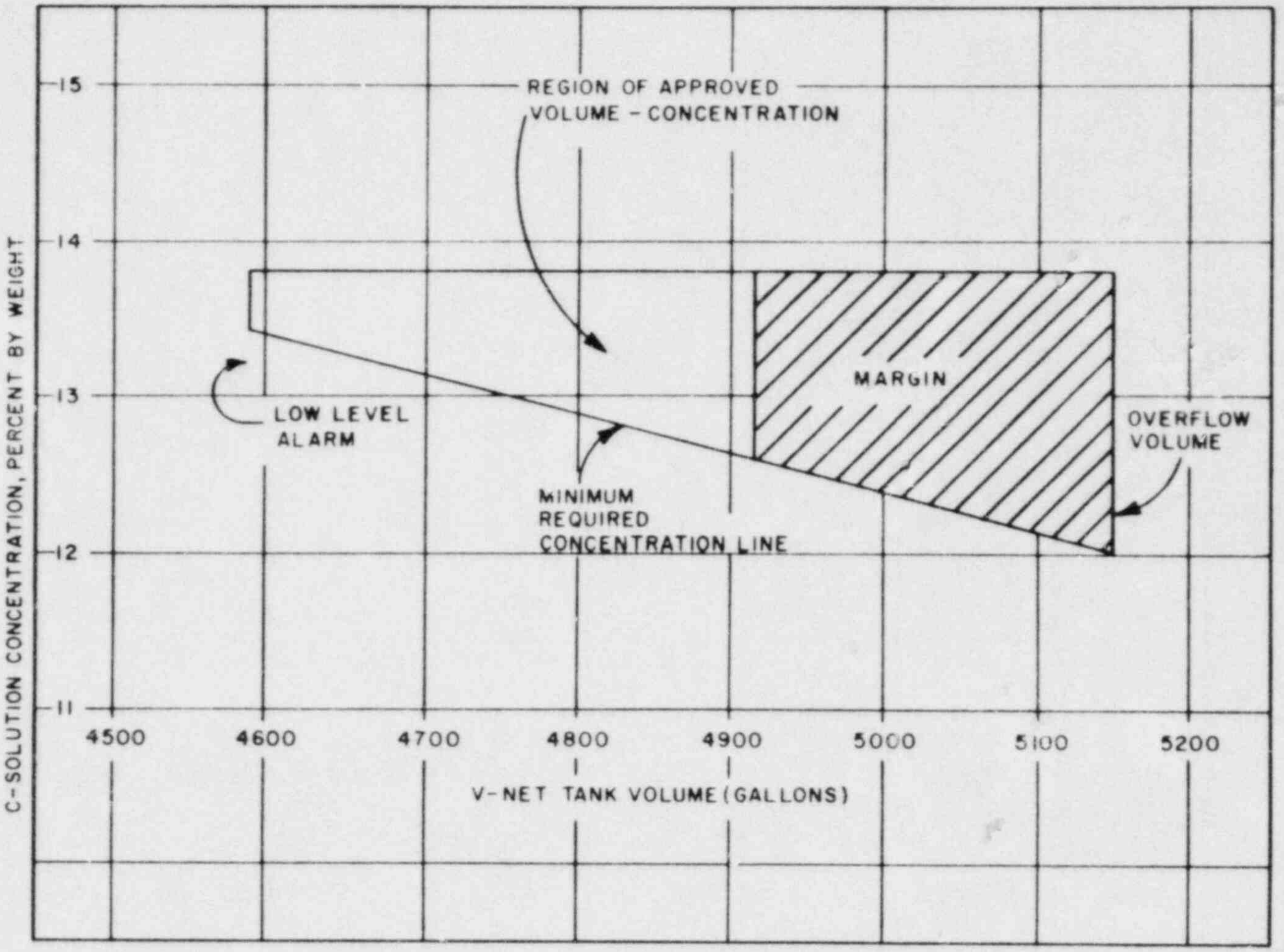
   3. **Demonstrating that all heat traced piping between the storage tank and the reactor vessel is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water.

   4. Demonstrating that the storage tank heaters are OPERABLE by verifying the expected temperature rise of 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 the 70°F limit.

**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.



SODIUM PENTABORATE VOLUME/CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

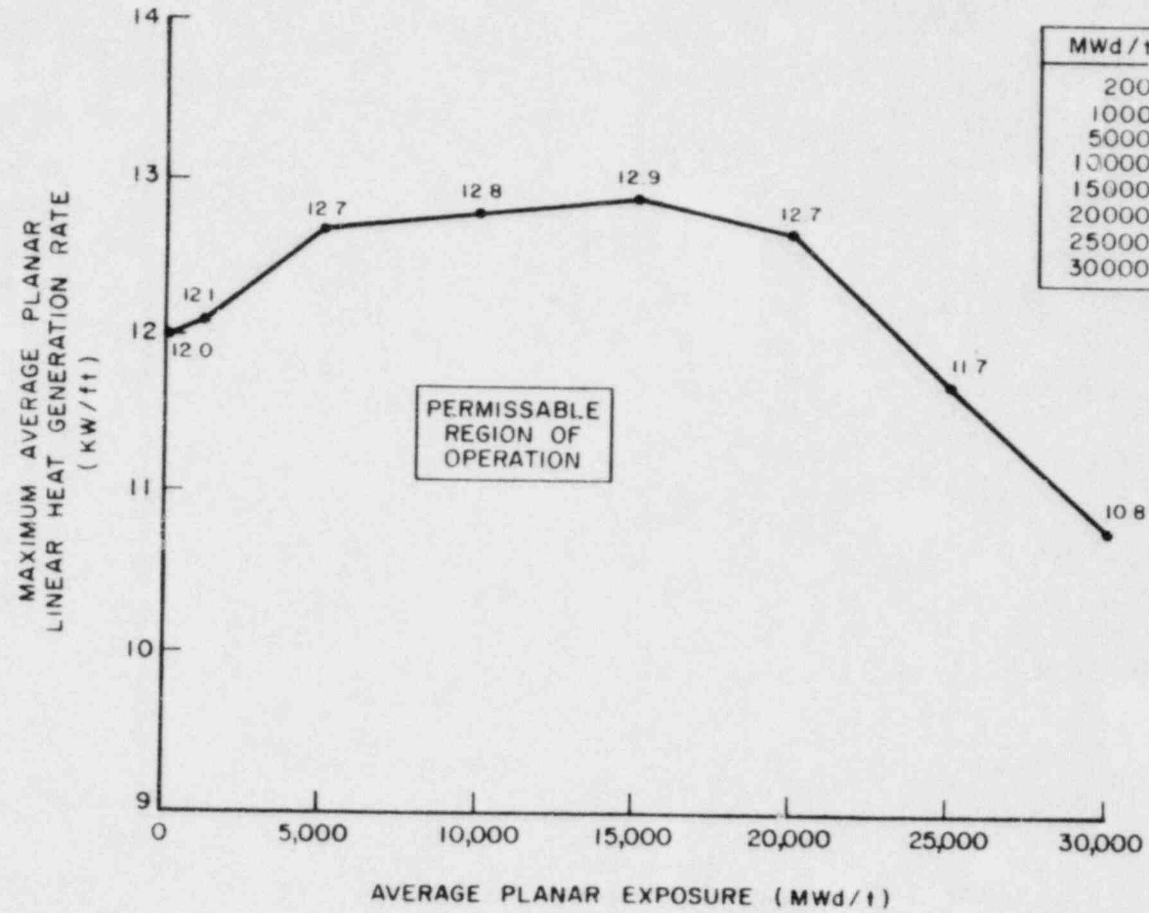
SURVEILLANCE REQUIREMENTS

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

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

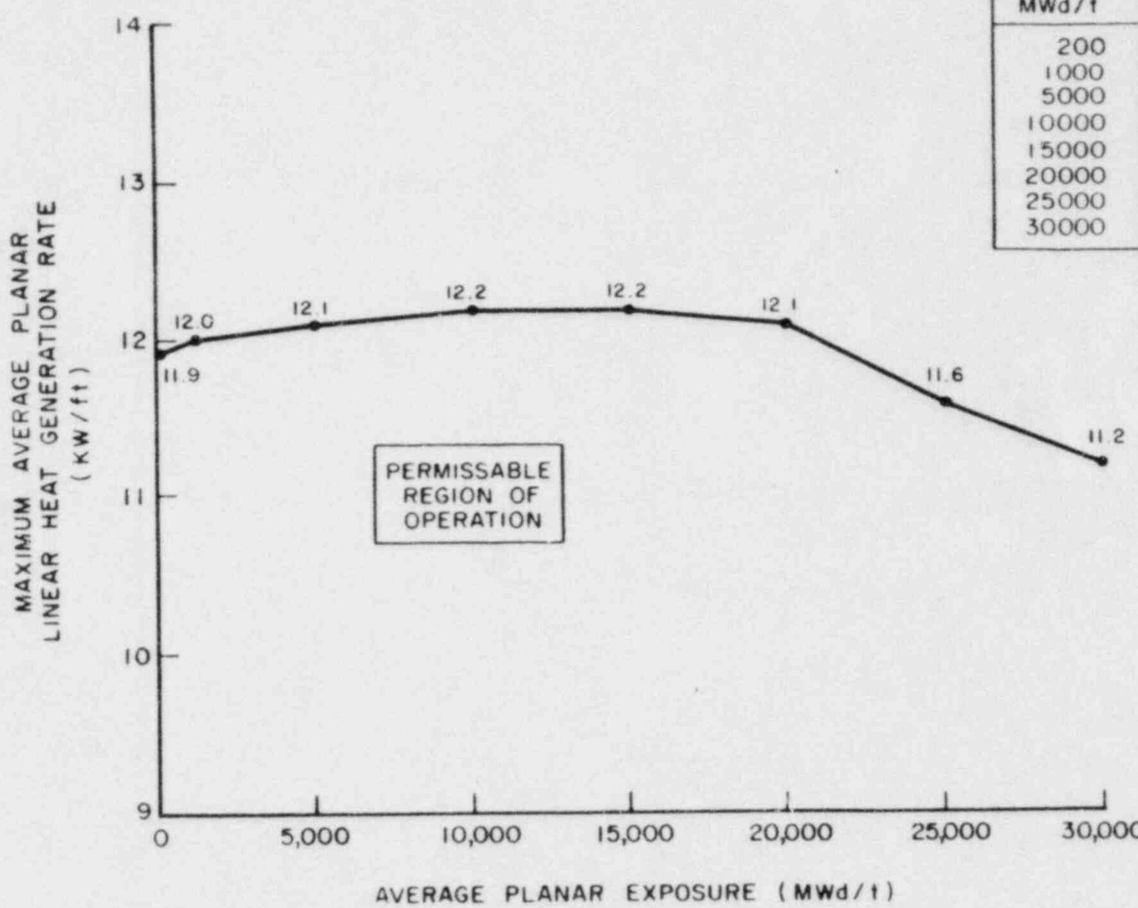
c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.



MWd/t	kW/ft
200	12.0
1000	12.1
5000	12.7
10000	12.8
15000	12.9
20000	12.7
25000	11.7
30000	10.8

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CI183

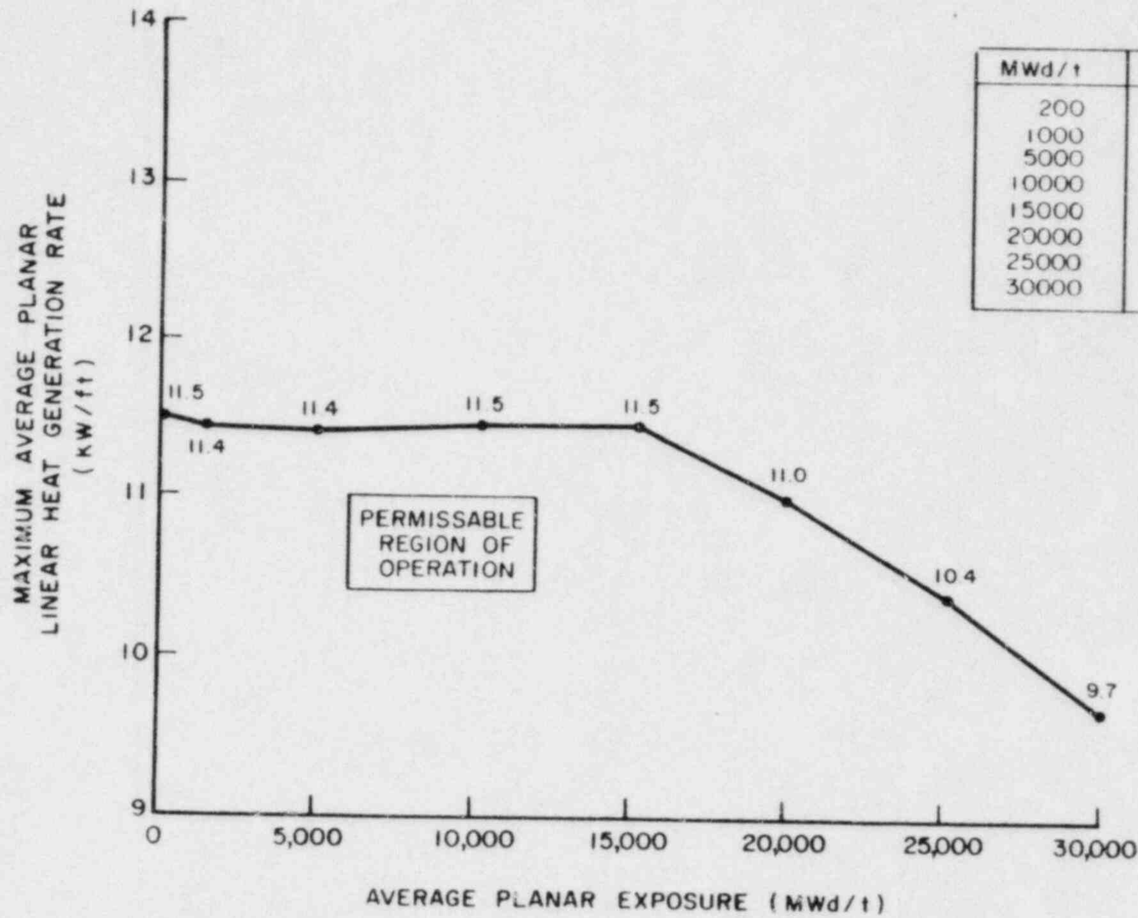
FIGURE 3.2.1-1



MWd/t	kW/ft
200	11.9
1000	12.0
5000	12.1
10000	12.2
15000	12.2
20000	12.1
25000	11.6
30000	11.2

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CI233

FIGURE 3.2.1-2



MWd/t	kW/ft
200	11.5
1000	11.4
5000	11.4
10000	11.5
15000	11.5
20000	11.0
25000	10.4
30000	9.7

MAXIMUM AVERAGE PLANAR LINEAR HEAT
GENERATION RATE (MAPLHGR) VERSUS
AVERAGE PLANAR EXPOSURE
INITIAL CORE FUEL TYPE 8CI711

FIGURE 3.2.1-3

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.66W + 51\%)T$	$S \leq (0.66W + 54\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

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 100 million lbs/hr, at 100% of RATED THERMAL POWER
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. 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 neutron flux-high scram trip setpoint and/or the flow biased neutron flux-high 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 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased neutron flux-high scram and flow biased neutron flux-high control rod block trip setpoints verified to be within the above limits or adjusted, or the ARPM gain readings shall be verified as indicated below, 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 MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

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 the MCPR limit shown in Figure 3.2.3-1 times the K_f shown in Figure 3.2.3-2, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 1.096 seconds, control rod average scram insertion time limit to notch 36 per Specification 3.1.3.3,

$$\tau_B = 0.852 + 1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} 0.06,$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 36 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

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

ACTION

a. 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.

b. With the main turbine bypass system inoperable per Specification 3.7.9, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within one hour, MCPR is determined to be equal to or greater than both $MCPR_f$ and $MCPR_p$ as shown in Figures 3.2.3-1 and 3.2.3-2 by the main turbine bypass inoperable curve.

4.2.3 MCPR, with:

a. $\tau = 1.0$ prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable 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.

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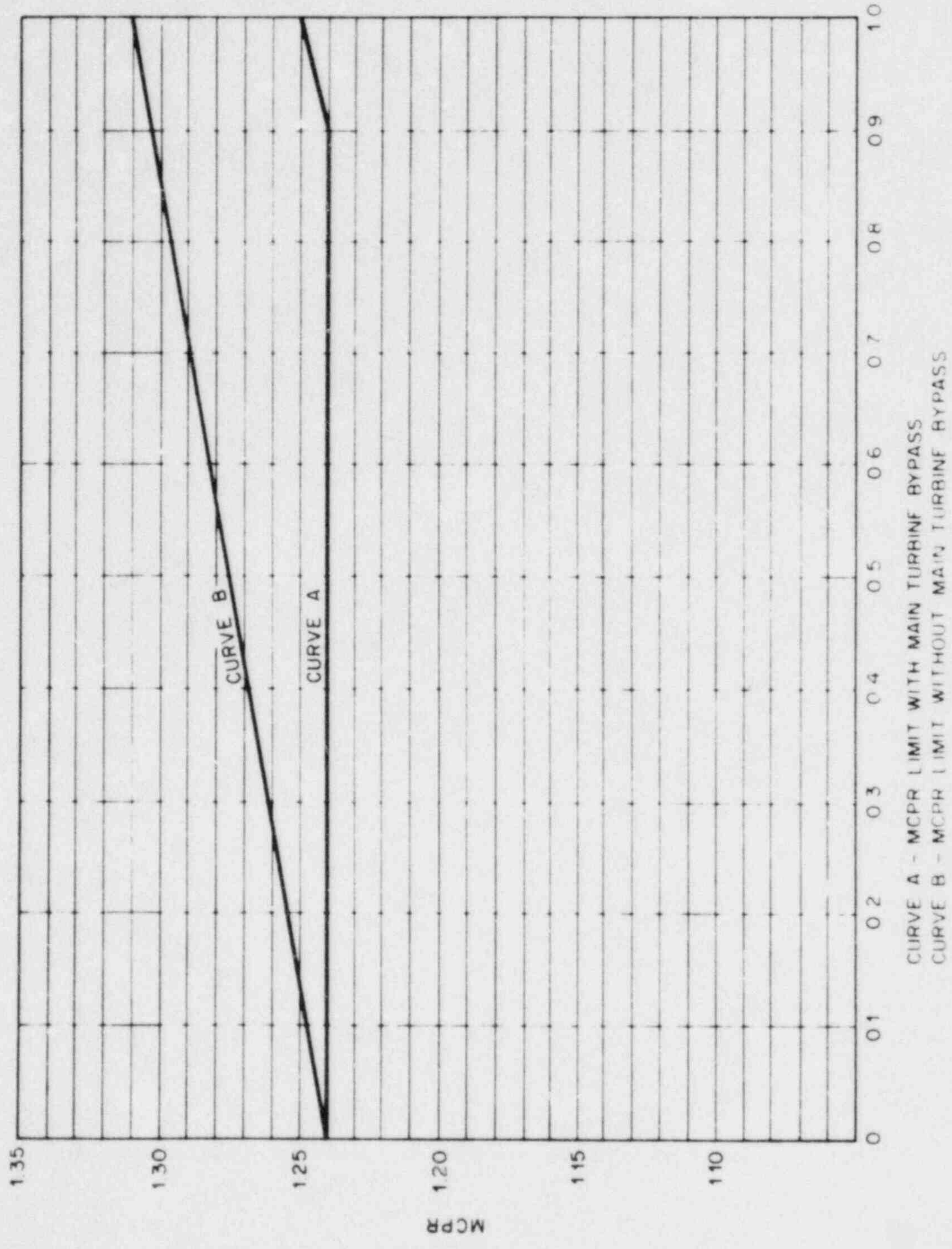
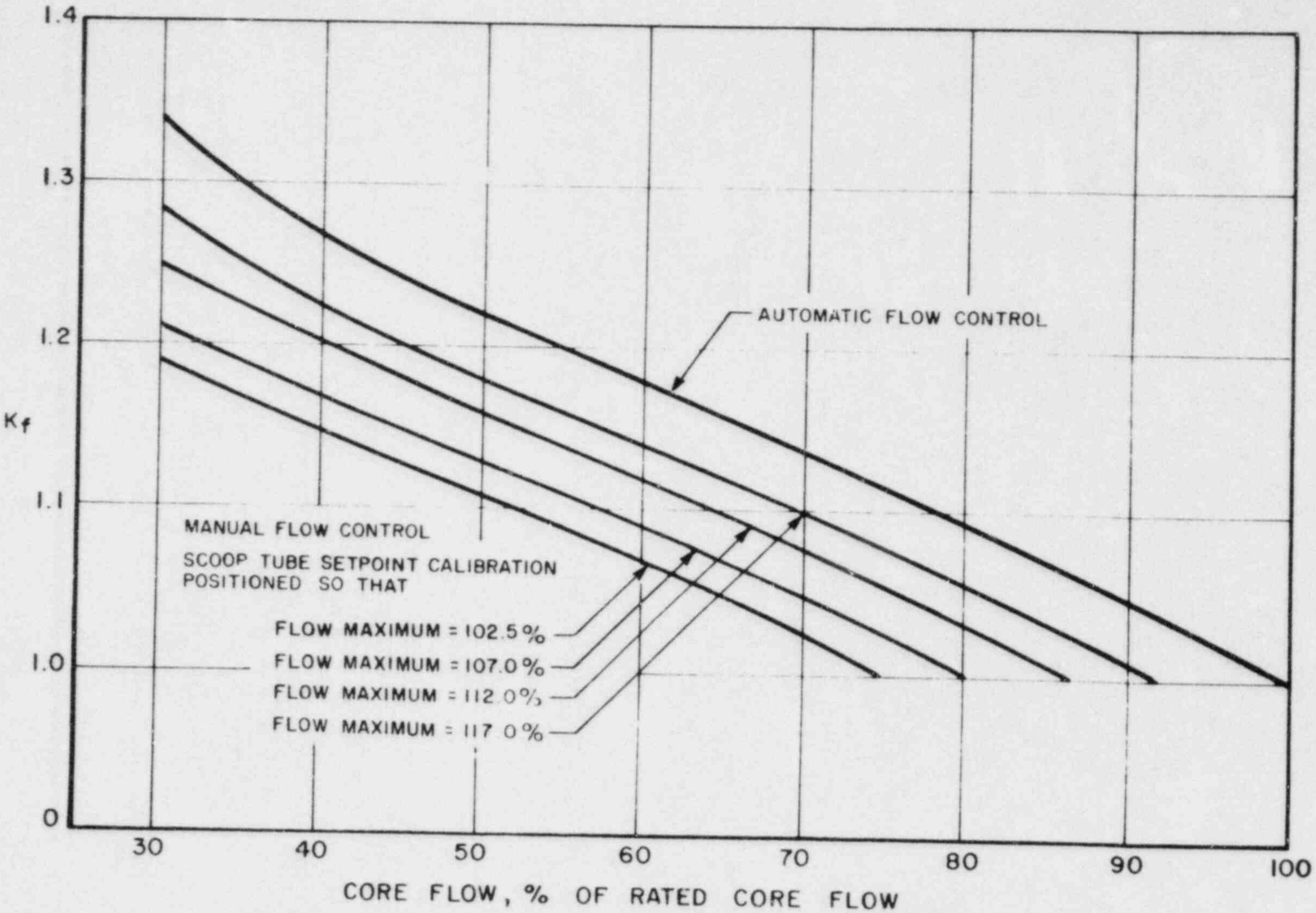


MINIMUM CRITICAL POWER RATIO (MCPR) VERSUS τ AT RATED FLOW

FIGURE 3.2.3-1



FLOW CORRECTION (K_f) FACTOR

FIGURE 3.2.3-2

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.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 1 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 1 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.

---

*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.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors[b]:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5(c)	3(d)	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3(d)	3
2. Average Power Range Monitor[e]:			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	5(c)	2(d)	3
b. Flow Biased Neutron Flux - High	1	2	4
c. Fixed Neutron Flux - High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5(c)	2(d)	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Low Water Level - Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. Main Steam Line Radiation - High	1, 2(f)	2	5
7. Drywell Pressure - High	1, 2(h)	2	1
8. Scram Discharge Volume Water Level - High			
a. Float Switches	1, 2	2	1
	5(i)	2	3
b. Level Transmitters	1, 2	2	1
	5(i)	2	3
9. Turbine Stop Valve - Closure	1(j)	4	6
10. Turbine Control Valve Fast Closure	1(j)	2	6
11. Reactor Mode Switch Shutdown Position	1, 2	1	1
	3, 4	1	7
	5	1	3
12. Manual Scram	1, 2	1	1
	3, 4	1	8
	5	1	9
13. Backup Manual Scram	1, 2	1	1
	3, 4	1	8
	5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

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

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

ACTION 3 - Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 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 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to ≤ 295 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 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 1 hour.

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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) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) 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.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

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

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

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

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

(j) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 295 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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 Neutron Flux - High	6 ± 1**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Low Water Level - Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	
a. Float Switch	NA
b. Level Transmitter	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure	$\leq$ 0.08***
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Backup 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.

**Including simulated thermal power time constant.

***Measured from deenergization of K-37 relay which inputs the turbine control valve closure signal to the RPS.

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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 FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b)	S/U[c], W	SA	2
	S	W	SA	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, 5
b. Flow Biased Neutron Flux - High	S,D[g]	S/U[c], W	W[d][e],SA,R[h]	1
c. Fixed Neutron Flux - High	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R	1, 2
4. Reactor Vessel Low Water Level - Level 3	S	M	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2[i]
7. Drywell Pressure - High	S	M	R	1, 2

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
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter	S	M	R	1, 2, 5(j)
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure	NA	M	NA	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5
13. Backup Manual Scram	NA	R	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 ½ decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least ½ decades 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 $\geq$ 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.

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

(g) Verify measured core flow to be less than or equal to established core flow at the existing loop drive flow.

(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

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

(j) 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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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.

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

SURVEILLANCE REQUIREMENTS

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

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. CONTAINMENT ISOLATION				
a. Reactor Vessel Low Water Level				
1) Level 3	4, 13, 15	2	1, 2, 3	20
2) Level 2	2, 10, 11, 16, 17, 18	2	1, 2, 3	20
	12, 14(b)	2	1, 2, 3 and *	24
3) Level 1	1	2	1, 2, 3	20
b. Drywell Pressure - High	2, 13, 15, 16, 17, 18	2	1, 2, 3	20
	12, 14(b)	2	1, 2, 3	24
c. Main Steam Line				
1) Radiation - High	1, 2(b)	2	1, 2, 3	21
2) Pressure - Low	1	2	1	22
3) Flow - High	1	2(c)	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2(c)	1, 2, 3	21
e. Condenser Pressure - High	1	2	1, 2**, 3**	21
f. Turbine Bldg. Area Temperature - High	1	2	1, 2, 3	21
g. Fuel Pool Ventilation Exhaust Radiation - High	14(b), 16	2	1, 2, 3, and *	24
h. Manual Initiation	1, 2, 4, 12, 13, 14(b) 15, 16, 17, 18	1/valve	1, 2, 3 and *	26

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
2. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	10, 11	1	1, 2, 3	23
b. Heat Exchanger/Pump Area Temperature - High	10, 11	1	1, 2, 3	23
c. Heat Exchanger/Pump Area Ventilation Δ Temp. - High	10, 11	1	1, 2, 3	23
d. SLCS Initiation	11	NA	1, 2, 3	23
e. Reactor Vessel Low Water Level - Level 2	10, 11(d)	2	1, 2, 3	23
f. NRHX Outlet Temperature - High	10, 11	2	1, 2, 3	23
g. Manual Initiation	10, 11	1/valve	1, 2, 3	26
3. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	8	1	1, 2, 3	23
b. RCIC Steam Supply Pressure - Low	8, 9	2	1, 2, 3	23
c. RCIC Turbine Exhaust Diaphragm Pressure - High	8	2	1, 2, 3	23
d. RCIC Equipment Room Temperature - High	8	1	1, 2, 3	23
e. Manual Initiation	8, 9	1/valve	1, 2, 3	26

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. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow - High	6	1	1, 2, 3	23
b. HPCI Steam Supply Pressure-Low	6, 7	2	1, 2, 3	23
c. HPCI Turbine Exhaust Diaphragm Pressure - High	6	2	1, 2, 3	23
d. HPCI Equipment Room Temperature - High	6	1	1, 2, 3	23
e. Manual Initiation	6, 7	1/valve	1, 2, 3	26
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Low Water Level - Level 3	$4^{(e)}$	2	1, 2, 3	25
b. Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	4	1	1, 2, 3	25
c. Manual Initiation	4	1/valve	1, 2, 3	26

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

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

ACTION 21 - 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 22 - Be in at least STARTUP within 6 hours.

ACTION 23 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

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

TABLE NOTATIONS

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

** May be bypassed under administrative control.

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel or trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition, for the HPCI system and RCIC system isolation, provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is OPERABLE and all required actuation instrumentation for that valve is OPERABLE, one channel may be placed in an inoperable status for up to 8 hours for required surveillance without placing the channel or trip system in the tripped condition.

(b) Also starts the standby gas treatment system.

(c) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.

(d) This level signal also actuates Groups 2, 14, 16, 17, and 18.

(e) This level signal also actuates Groups 13 and 15.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. CONTAINMENT ISOLATION		
a. Reactor Vessel Low Water Level		
1) Level 3	≥ 173.4 inches*	≥ 171.9 inches
2) Level 2	≥ 110.8 inches*	≥ 103.8 inches
3) Level 1	≥ 31.8 inches	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
2) Pressure - Low	≥ 756 psig	≥ 736 psig
3) Flow - High	≤ 137.9% of rated flow/109.0 psid	≤ 139.5% of rated flow/112.0 psid
d. Main Steam Line Tunnel Temperature - High	≤ 200°F**	≤ 206°F**
e. Condenser Pressure - High	≤ 6.85 psia	≤ 7.05 psia
f. Turbine Bldg. Area Temperature - High	≤ 200°F	≤ 206°F
g. Fuel Pool Ventilation Exhaust Radiation - High	≤ 10 mR/hr**	≤ 15 mR/hr**
h. Manual Initiation	NA	NA

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 59 gpm	≤ 66 gpm
b. Heat Exchanger/Pump Area Temperature - High	≤ 175°F**	≤ 183°F**
c. Heat Exchanger/Pump Area Ventilation Δ Temperature - High	≤ 50°F**	≤ 53°F**
d. SLCS Initiation	NA	NA
e. Reactor Vessel Low Water Level - Level 2	≥ 110.8 inches*	≥ 103.8 inches
f. NRHX Outlet Temperature - High	≤ 140°F	≤ 150°F
g. Manual Initiation	NA	NA
3. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 104.5" H_2O/99,000 lbm/hr	≤ 116.3" H_2O/104,500 lbm/hr
b. RCIC Steam Supply Pressure - Low	≥ 62 psig	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. RCIC Equipment Room Temperature - High	≤ 150°F**	≤ 160°F**
e. Manual Initiation	NA	NA

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TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High	≤ 425.7 inches H_2O/553,500 lbm/hr	≤ 443.5 inches H_2O/564,900 lbm/hr
b. HPCI Steam Supply Pressure - Low	≥ 100 psig	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. HPCI Equipment Room Temperature - High	≤ 150°F**	≤ 162°F**
e. Manual Initiation	NA	NA
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches
b. Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	≤ 89.5 psig***	≤ 95.5 psig***
c. Manual Initiation	NA	NA

*Above TAF. 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.

***Represents steam dome pressure; actual trip setpoint is corrected for cold water head with reactor vessel flooded.

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TABLE 3.3.2-3

ISOLATION ACTUATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. CONTAINMENT ISOLATION	
a. Reactor Vessel Low Water Level	
1) Level 3	$\leq 13^{(a)}$
2) Level 2	$\leq 13^{(a)**}$
3) Level 1	≤ 1.0*/$\leq 13^{(a)**}$
b. Drywell Pressure - High	$\leq 13^{(a)}$
c. Main Steam Line	
1) Radiation - High(b)	$\leq 13^{(a)**}$
2) Pressure - Low	$\leq 13^{(a)**}$
3) Flow - High	$\leq 13^{(a)**}$
d. Main Steam Line Tunnel Temperature - High	NA
e. Condenser Pressure - High	NA
f. Turbine Bldg. Area Temperature - High	NA
g. Fuel Pool Ventilation Exhaust Radiation - High(b)	$\leq 13^{(a)}$
h. Manual Initiation	NA
2. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	NA##
b. Heat Exchanger/Pump Area Temperature - High	NA
c. Heat Exchanger/Pump Area Ventilation Temperature ΔT - High	NA
d. SLCS Initiation	NA
e. Reactor Vessel Low Water Level - Level 2	$\leq 13^{(a)}$
f. NRHX Outlet Temperature - High	NA
g. Manual Initiation	NA
3. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	NA###
b. RCIC Steam Supply Pressure - Low	$\leq 13^{(a)}$
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Temperature - High	NA
e. Manual Initiation	NA

TABLE 3.3.2-3 (Continued)

ISOLATION ACTUATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION		RESPONSE TIME (Seconds)#
4.	HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION	
a.	HPCI Steam Flow - High	NA####
b.	HPCI Steam Supply Pressure - Low	$\leq 13^{(a)}$
c.	HPCI Turbine Exhaust Diaphragm Pressure - High	NA
d.	HPCI Equipment Room Temperature - High	NA
e.	Manual Initiation	NA
5.	RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a.	Reactor Vessel Low Water Level - Level 3	NA
b.	Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	NA
c.	Manual Initiation	NA

(a) The isolation system instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME. Isolation system instrumentation response time specified includes 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 for MSIVs.

**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 Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

##With time delay of 45 seconds.

###With time delay of 13 + 0, -1 seconds.

####With time delay of 13 + 0, -1 seconds.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. CONTAINMENT ISOLATION				
a. Reactor Vessel Low Water Level-				
1) Level 3	S	M	R	1, 2, 3
2) Level 2	S	M	R	1, 2, 3, and *
3) Level 1	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	M	R	1, 2, 3
2) Pressure - Low	S	M	R	1
3) Flow - High	S	M	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
e. Condenser Pressure - High	S	M	R	1, 2**, 3**
f. Turbine Bldg. Area Temperature - High	S	M	R	1, 2, 3
g. Fuel Pool Ventilation Exhaust Radiation - High	S	M	R	1, 2, 3, and *
h. Manual Initiation	NA	R	NA	1, 2, 3, and *

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
2. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Heat Exchanger/Pump Area Temperature - High	S	M	R	1, 2, 3
c. Heat Exchanger/Pump Area Ventilation Δ Temperature - High	S	M	R	1, 2, 3
d. SLCS Initiation	NA	M	NA	1, 2, 3
e. Reactor Vessel Low Water Level - Level 2	S	M	R	1, 2, 3
f. NRHX Outlet Temperature - High	S	M	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3
3. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	M	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	M	R	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. RCIC Equipment Room Temperature - High	S	M	R	1, 2, 3
e. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION				
a. HPCI Steam Line Flow - High	S	M	R	1, 2, 3
b. HPCI Steam Supply Pressure - Low	S	M	R	1, 2, 3
c. HPCI Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
d. HPCI Equipment Room Temperature - High	S	M	R	1, 2, 3
e. Manual Initiation	NA	R	NA	1, 2, 3
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Low Water Level - Level 3	S	M	R	1, 2, 3
b. Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	S	M	R	1, 2, 3
c. Manual Initiation	NA	R	NA	1, 2, 3

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

** May be bypassed under administrative control.

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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.

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one 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 SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Low Water Level - Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. Reactor Steam Dome Pressure - Low	2	1, 2, 3	30
(Injection Permissive)	2	4*, 5*	30
d. Manual Initiation	1##	1, 2, 3, 4*, 5*	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Low Water Level - Level 1	2	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. Reactor Steam Dome Pressure - Low (Valve	2	1, 2, 3	30
Permissive)	2	4*, 5*	30
d. Reactor Vessel Low Water Level - Level 2 (Loop Select Logic)	2	1, 2, 3, 4*, 5*	30
e. Reactor Steam Dome Pressure - Low (Break Detection Logic)	2	1, 2, 3, 4*, 5*	30
f. Riser Differential Pressure - High (Break Detection)	2	1, 2, 3	30
g. Recirculation Pump Differential Pressure - High (Break Detection)	2	1, 2, 3	30
h. Manual Initiation	1##	1, 2, 3, 4*, 5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Low Water Level - Level 2	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. Condensate Storage Tank Level - Low	2(c)	1, 2, 3	34
d. Suppression Pool Water Level - High	2(d)	1, 2, 3	34
e. Reactor Vessel High Water Level - Level 8	2(e)	1, 2, 3	30
f. Manual Initiation	1##	1, 2, 3	33

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
4. AUTOMATIC DEPRESSURIZATION SYSTEM#			
a. Reactor Vessel Low Water Level - Level 1	2	1, 2, 3	30
b. Drywell Pressure - High	2	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31
e. RHR LPCI Mode Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31
f. Reactor Vessel Low Water Level - Level 3 (Permissive)	1	1, 2, 3	31
g. Manual Initiation	1/valve	1, 2, 3	33

TRIP FUNCTION	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
5. LOSS OF POWER					
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	2/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	35
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	2/bus	1/bus	1/bus	1, 2, 3, 4**, 5**	35

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the associated emergency diesel generators.
(c) One trip system. Provides signals to HPCI and RCIC suction valves.
(d) One trip system. Provides signal to HPCI pump suction valves only.
(e) On 2 out of 2 logic, provides a signal to trip the HPCI turbine.
* When the system is required to be OPERABLE per Specification 3.5.2.
** Required when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is less than an equal to 150 psig.
\#\# Individual component controls.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION STATEMENTS

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

a. For one trip system, place that trip system in the tripped condition within 1 hour* or declare the associated ECCS inoperable.

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

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

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place the inoperable channel in the tripped condition within 1 hour.

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

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 1 hour*, align the HPCI system to take suction from the suppression pool, or declare the HPCI system inoperable.

ACTION 35 - With the number of OPERABLE channels:

a. One less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 72 hours or 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.

b. Less than the Minimum Channels OPERABLE requirement, declare the associated 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
1. CORE SPRAY SYSTEM		
a. Reactor Vessel Low Water Level - Level 1	≥ 31.8 inches*	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Steam Dome Pressure - Low	≥ 461 psig, decreasing	≥ 441 psig, decreasing
d. Manual Initiation	NA	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM		
a. Reactor Vessel Low Water Level - Level 1	≥ 31.8 inches*	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Steam Dome Pressure - Low	≥ 461 psig, decreasing	≥ 441 psig, decreasing
d. Reactor Vessel Low Water Level - Level 2	≥ 110.8 inches*	≥ 103.8 inches
e. Reactor Steam Dome Pressure - Low	≥ 906 psig, decreasing	≥ 886 psig, decreasing
f. Riser Differential Pressure - High	≤ 0.627 psid	≤ 0.927 psid
g. Recirculation Pump Differential Pressure - High	≤ 1.627 psid	≤ 1.927 psid
h. Manual Initiation	NA	NA
3. HIGH PRESSURE COOLANT INJECTION SYSTEM		
a. Reactor Vessel Low Water Level - Level 2	≥ 110.8 inches*	≥ 103.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Condensate Storage Tank Level - Low	≥ 3 inches (27 inches above tank bottom)	≥ 0 inches (24 inches above tank bottom)
d. Suppression Pool Water Level - High	≥ 2.0 inches**	≥ 5.0 inches**
e. Reactor Vessel High Water Level - Level 8	≤ 214 inches*	≤ 219 inches
f. 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
4. AUTOMATIC DEPRESSURIZATION SYSTEM		
a. Reactor Vessel Low Water Level - Level 1	≥ 31.8 inches*	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. ADS Timer	≤ 105 seconds	≤ 117 seconds
d. Core Spray Pump Discharge Pressure - High	≥ 145 psig, increasing	≥ 125 psig, increasing
e. RHR LPCI Mode Pump Discharge Pressure-High	≥ 125 psig, increasing	≥ 115 psig, increasing
f. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches
g. Manual Initiation	NA	NA
5. LOSS OF POWER	Division 1	
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) (Division 1 and Division 2)	a. 4.16 kV Basis - 3033 volts	3033 ± 60.7 volts
	b. 120 V Basis - 87.5 volts	87.5 ± 1.75 volts
	c. 2 sec time delay	2.0 ± 0.1 sec time delay
	Division 2	
	a. 4.16 kV Basis - 3078 volts	3078 ± 61.6 volts
	b. 120 V Basis - 88.8 volts	88.8 ± 1.78 volts
	c. 2 sec time delay	2.0 ± 0.1 sec time delay
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) (Division 1 and Division 2)	Division 1	
	a. 4.16 kV Basis - 3702 volts	3702 ± 74.0 volts
	b. 120 V Basis - 106.8 volts	106.8 ± 2.14 volts
	c. 19.7 sec time delay	19.7 ± 1.0 sec time delay
	Division 2	
	a. 4.16 kV Basis - 3432 volts	3432 ± 68.6 volts
	b. 120 V Basis - 99.0 volts	99.0 ± 2.0 volts
	c. 21.4 sec time delay	21.4 ± 1.07 sec time delay

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

** Suppression pool water level instrument zero is 14'8" above bottom of torus at elevation 556'0".

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TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

TRIP FUNCTION	RESPONSE TIME (Seconds)
1. CORE SPRAY SYSTEM	
a. Reactor Vessel Low Water Level - Level 1.	≤30
b. Drywell Pressure-High	≤30
c. Reactor Steam Dome Pressure-Low	≤30
d. Manual Initiation	NA
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
a. Reactor Vessel Low Water Level - Level 1	≤ 43
b. Drywell Pressure - High	≤ 43
c. Reactor Steam Dome Pressure - Low	≤ 43
d. Reactor Vessel Low Water Level - Level 2	NA
e. Reactor Steam Dome Pressure - Low	NA
f. Riser Differential Pressure - High	NA
g. Recirculation Pump Differential Pressure - High	NA
h. Manual Initiation	NA
3. HIGH PRESSURE COOLANT INJECTION SYSTEM	
a. Reactor Vessel Low Water Level - Level 2	≤ 30
b. Drywell Pressure - High	≤ 30
c. Condensate Storage Tank Level-Low	NA
d. Reactor Vessel Water Level-High, Level 8	NA
e. Suppression Pool Water Level-High	NA
f Manual Initiation	NA
4. AUTOMATIC DEPRESSURIZATION SYSTEM	
a. Reactor Vessel Low Water Level - Level 1	NA
b. Drywell Pressure-High	NA
c. ADS Timer	NA
d. Core Spray Pump Discharge Pressure-High	NA
e. RHR LPCI Mode Pump Discharge Pressure-High	NA
f. Reactor Vessel Low Water Level - Level 3	NA
g. Manual Initiation	NA
5. LOSS OF POWER	
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	NA
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	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
1. CORE SPRAY SYSTEM				
a. Reactor Vessel Low Water Level - Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM				
a. Reactor Vessel Low Water Level - Level 1	S	M	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
d. Reactor Vessel Low Water Level - Level 2	S	M	R	1, 2, 3, 4*, 5*
e. Reactor Steam Dome Pressure - Low	S	M	R	1, 2, 3, 4*, 5*
f. Riser Differential Pressure - High	S	M	R	1, 2, 3
g. Recirculation Pump Differential Pressure - High	S	M	R	1, 2, 3
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
3. HIGH PRESSURE COOLANT INJECTION SYSTEM				
a. Reactor Vessel Low Water Level - Level 2	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Condensate Storage Tank Level - Low	S	M	R	1, 2, 3
d. Suppression Pool Water Level - High	S	M	R	1, 2, 3
e. Reactor Vessel High Water Level - Level 8	S	M	R	1, 2, 3
f. 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
4. AUTOMATIC DEPRESSURIZATION SYSTEM#				
a. Reactor Vessel Low Water Level - Level 1	S	M	R	1, 2, 3
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. ADS Timer	NA	R	R	1, 2, 3
d. Core Spray Pump Discharge Pressure - High	S	M	R	1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High	S	M	R	1, 2, 3
f. Reactor Vessel Low Water Level - Level 3	S	M	R	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage) (Division 1 and Division 2)	NA	M	R	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage) (Division 1)	NA	M	R	1, 2, 3, 4**, 5**

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

3/4.3.4 ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

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 3.3.4-2, 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 trip function in one trip system, restore the inoperable channel to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1 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.

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

TABLE 3.3.4-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Reactor Vessel Low Water Level - Level 2	2
2. Reactor Vessel Pressure-High	2

*One channel in one trip system 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 that all other channels are OPERABLE.

TABLE 3.3.4-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Low Water Level - Level 2	$\geq$ 110.8 inches*	$\geq$ 103.8 inches
2. Reactor Vessel Pressure-High	$\leq$ 113[illegible] psig	$\leq$ 1153 psig

*See Bases Figure B3/4 3-1.

TABLE 4.3.4-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
a. Reactor Vessel Low Water Level - Level 2	2	50
b. Reactor Vessel High Water Level - Level 8	2[(b)]	50
c. Condensate Storage Tank Water Level - Low	2[(c)]	51
d. Manual Initiation	1[(d)]	52

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

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

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

(d) One trip system with one channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM

ACTION STATEMENTS

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

a. For one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition within 1 hour or declare the RCIC system inoperable.

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

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within 1 hour* or align RCIC to take suction from the suppression pool or declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System 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.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Low Water Level - Level 2	$\geq$ 110.8 inches*	$\geq$ 103.8 inches
b. Reactor Vessel High Water Level - Level 8	$\leq$ 214 inches*	$\leq$ 219 inches
c. Condensate Storage Tank Level - Low	$\geq$ 3 inches (27 inches above tank bottom)	$\geq$ 0 inches (24 inches above tank bottom)
d. Manual Initiation	NA	NA

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

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 Low Water Level - Level 2	S	M	R
b. Reactor Vessel High Water Level - Level 8	S	M	R
c. Condensate Storage Tank Level - Low	S	M	R
d. Manual Initiation	NA	R	NA

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.

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

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

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TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION STATEMENTS

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

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

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within 1 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 1 hour.

TABLE NOTATIONS

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

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

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

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

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

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ 0.66 W + 40%*	$\leq$ 0.66 W + 43%*
b. Inoperative	NA	NA
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - High	$\leq$ 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, Setdown	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq 1.0 \times 10^5$ cps	$\leq 1.6 \times 10^5$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ 3 cps	$\geq$ 2 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ 108/125 divisions of full scale	$\leq$ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ 5/125 divisions of full scale	$\geq$ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	$\leq$ 70 gallons**	$\leq$ 90 gallons**
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ 108/125% of rated flow	$\leq$ 111/125% of rated flow
b. Inoperative	NA	NA
c. Comparator	$\leq$ 10% flow deviation	$\leq$ 11% flow deviation

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

**Volume is measured from closed drain valve C11-F011.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[(b)],M	Q	1*
b. Inoperative	NA	S/U[(b)],M	NA	1*
c. Downscale	NA	S/U[(b)],M	Q	1*
2. APRM				
a. Flow Biased Neutron Flux - High	S	S/U[(b)],M	SA	1
b. Inoperative	NA	S/U[(b)],M	NA	1, 2, 5
c. Downscale	S	S/U[(b)],M	SA	1
d. Neutron Flux - Upscale, Setdown	S	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	S	S/U[(b)],W	SA	2***, 5
c. Inoperative	NA	S/U[(b)],W	NA	2***, 5
d. Downscale	S	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	S	S/U[(b)],W	SA	2, 5
c. Inoperative	NA	S/U[(b)],W	NA	2, 5
d. Downscale	S	S/U[(b)],W	SA	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypass	NA	R	NA	2, 5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[(b)],M	Q	1
b. Inoperative	NA	S/U[(b)],M	NA	1
c. Comparator	NA	S/U[(b)],M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

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

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

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

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

*** With IRMs on Range 2 or less.

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Main Control Room Ventilation Radiation Monitor	1/intake	1,2,3,5 and *	$\leq$ 5 mR/hr	70
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Vault	1	#	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr(a)	71
2) Fuel Storage Pool	1	##	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr(a)	71
b. Control Room Direct Radiation Monitor	1	At all times	$\leq$ 0.5 mR/hr(a)	71

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

*When irradiated fuel is being handled in the secondary containment.

#With fuel in the new fuel vault.

##With fuel in the fuel storage pool.

(a)Alarm only.

ACTION STATEMENTS

ACTION 70 -

a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the recirculation mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the recirculation mode of operation within 1 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.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Main Control Room Ventilation Radiation Monitor	S	M	R	1, 2, 3, 5, and *
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Vault	S	M	R	#
2) Fuel Storage Pool	S	M	R	##
b. Control Room Direct Radiation Monitor	S	M	R	At all times

#With fuel in the new fuel vault.

##With fuel in the fuel storage pool.

*When irradiated fuel is being handled in the secondary containment.

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SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.3.7.2.1 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 0.01 g 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. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum, and resultant effect upon unit features important to safety.

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TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Active Triaxial System		
a. Active Triaxial Accelerometers		
1) HPCI Room	±1 g	1
2) Base of RPV Pedestal, In Drywell	±1 g	1
b. Active Seismic Recording System*		
1) Relay Room, Auxiliary Building	NA	1**
c. Active Seismic Playback System		
1) Relay Room, Auxiliary Building	NA	NA
2. Passive Triaxial Peak Shock Recorders		
a. HPCI Room	***	1
b. Relay Room, Auxiliary Building	***	1
c. Refuel Floor, Reactor Building	***	1
d. Diesel Generator Room, RHR Complex	***	1
e. Pump Room, RHR Complex	***	1
f. Cooling Tower, RHR Complex	***	1

*Including seismic trigger.

**With reactor control room annunciation.

***Each passive accelerometer has 12 reeds, each monitoring a different frequency. The frequencies correspond to varying accelerations. The widest range is ± 90 g.

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Active Triaxial System			
a. Active Triaxial Accelerometers			
1) HPCI Room	NA	SA	R
2) Base of RPV Pedestal, In Drywell	NA	SA	R
b. Active Seismic Recording System*			
1) Relay Room, Auxiliary Building**	M(a)	SA	R
c. Active Seismic Playback System			
1) Relay Room, Auxiliary Building	M	SA	R
2. Passive Triaxial Peak Shock Recorders			
a. HPCI Room	NA	NA	R
b. Relay Room, Auxiliary Building	NA	NA	R
c. Refuel Floor, Reactor Building	NA	NA	R
d. Diesel Generator Room, RHR Complex	NA	NA	R
e. Pump Room, RHR Complex	NA	NA	R
f. Cooling Tower, RHR Complex	NA	NA	R

*Including seismic trigger.

**With reactor control room annunciation.

(a)Except seismic trigger.

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 a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
a. Wind Speed	
1. Elev. 10 meters	1
2. Elev. 60 meters	1
b. Wind Direction	
1. Elev. 10 meters	1
2. Elev. 60 meters	1
c. Air Temperature Difference	
1. Elev. 10/60 meters	1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 10 meters	D	SA
2. Elev. 60 meters	D	SA
b. Wind Direction		
1. Elev. 10 meters	D	SA
2. Elev. 60 meters	D	SA
c. Air Temperature Difference		
1. Elev. 10/60 meters	D	SA

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REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With 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. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	REMOTE SHUTDOWN PANEL DIVISION	MINIMUM INSTRUMENTS OPERABLE
1. Reactor Vessel Pressure	I, II	1/panel
2. Reactor Vessel Water Level	I, II	1/panel
3. Safety/Relief Valve Energization, 2 valves	I	1/valve
4. Suppression Chamber Water Temperature	I, II	1/panel
5. Drywell Pressure	I, II	1/panel
6. RHR Division I Hx Discharge Flow	I	1
7. RCIC Flow	I	1
8. HPCI Flow	II	1

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 Energization	M	NA
4. Suppression Chamber Water Temperature	M	R
5. Drywell Pressure	M	R
6. RHR Division I Hx Discharge Flow	M	R
7. RCIC Flow	M	R
8. HPCI Flow	M	R

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 OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	80
2. Reactor Vessel Water Level, Fuel Zone	2	1	1, 2	80
3. Suppression Chamber Water Level	2	1	1, 2	80
4. Suppression Chamber Water Temperature	2	1	1, 2	80
5. Suppression Chamber Air Temperature	2	1	1, 2	80
6. Suppression Chamber Pressure	2	1	1, 2	80
7. Drywell Pressure, Wide Range	2	1	1, 2	80
8. Drywell Air Temperature	2	1	1, 2	80
9. Drywell Oxygen Concentration	2	1	1, 2	80
10. Drywell Hydrogen Concentration	2	1	1, 2	80
11. Safety/Relief Valve Position Indicators	2*/valve	1*/valve	1, 2	80
12. Containment High Range Radiation Monitor	2	1	1, 2, 3	81
13. Standby Gas Treatment System Radiation Monitors**	2	1	1, 2, 3	81

*Pressure switches

**High (accident) range noble gas monitors

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TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

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

TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

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

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, Fuel Zone	M	R	1, 2
3. Suppression Chamber Water Level	M	R	1, 2
4. Suppression Chamber Water Temperature	M	R	1, 2
5. Suppression Chamber Air Temperature	M	R	1, 2
6. Suppression Chamber Pressure	M	R	1, 2
7. Drywell Pressure, Wide Range	M	R	1, 2
8. Drywell Air Temperature	M	R	1, 2
9. Drywell Oxygen Concentration	M	R	1, 2
10. Drywell Hydrogen Concentration	M	Q*	1, 2
11. Safety/Relief Valve Position Indicators	M	R	1, 2
12. Containment High Range Radiation Monitor	M	R**	1, 2, 3
13. Standby Gas Treatment System Radiation Monitors***	M	R	1, 2, 3

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

**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 (accident) range noble gas 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 CONDITIONS 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 inserted in the core and lock the reactor mode switch in the Shutdown position within 1 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 signal-to-noise ratio is $\geq$ 2. Otherwise, 3 cps.

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

b. Indexing equipment to allow all five 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, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8 Two independent chlorine detectors shall be OPERABLE with their trip setpoints adjusted to actuate at chlorine concentration of less than or equal to 5 ppm.

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

a. With one chlorine detector inoperable, restore the inoperable detector to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain isolation of all control room emergency intakes by placing the HVAC system in the chlorine mode of operation.

b. With both chlorine detectors inoperable, within 1 hour initiate and maintain isolation of all control room emergency intakes by placing the HVAC system in the chlorine mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.8 Each of the above required chlorine detectors 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.

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FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.9 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3.7.9-1 shall be OPERABLE.

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

ACTION:

a. With the number of OPERABLE fire detection instruments in one or more zones:

   1. Less than, but more than one-half of, the Total Number of Instruments shown in Table 3.3.7.9-1 for Function A, restore the inoperable Function A instrument(s) to OPERABLE status within 14 days or within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect 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 4.6.1.7.

   2. One less than the Total Number of Instruments shown in Table 3.3.7.9-1 for Function B, or one-half or less of the Total Number of Instruments shown in Table 3.3.7.9-1 for Function A, or with any two or more adjacent instruments inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then 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 4.6.1.7.

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

SURVEILLANCE REQUIREMENTS

4.3.7.9.1 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.

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

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

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	FIRE DETECTION ZONE	TOTAL NUMBERS OF INSTRUMENTS* IONIZATION (x/y)	PHOTOELECTRIC (x/y)	FIXED THERMAL (x/y)
a. Reactor Building				
1. Torus Area	1	8/0		
2. NW Corner Rooms, RHR Pump	2	3/0		
3. SW Corner Rooms, RHR Pump	3	3/0		
4. SE Corner Rooms, CRD HPCI	4	6/0		
5. NE Corner Rooms, RCIC	5	3/0		
6. First Floor	7	10/0		8/0
7. EECW System Area Second Floor	10	10/0		
8. Third Floor	15	14/0		
9. Fourth Floor	17	8/0		2/0
10. Refueling Area, Fifth Floor	17	9/0		
b. Auxiliary Building				
1. Basement, N Control Air Equipment	4	5/0		
2. Corridors, 562', 563'	5	2/0	2/0	
3. First Floor Mezzanine, Cable Trays, 583', 603'	6	12/0		
4. Switchgear Room, Corridor Area Second Floor	9	9/0		
5. Cable Tunnel	9	10/0		
6. Cable Tray Area Second Floor Mezzanine	9A	0/22		
7. DC/MCC Room, Third Floor	14	0/10		
8. Switchgear, Battery and M-G Rooms, Third Floor	14	12/0		
9. Fourth Floor	16	5/0		
10. Fifth Floor	16	20/0		

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TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	FIRE DETECTION ZONE	TOTAL NUMBER OF INSTRUMENTS* IONIZATION (x/y)	PHOTOELECTRIC (x/y)	FIXED THERMAL (x/y)
c. Control Center				
1. Relay Room	8	0/27		
2. Cable Spreading Room	11	0/28		
3. Control Room	12	45/0	4/0	2/0
4. Computer Room	13	0/13		
5. Computer Room above Drop ceiling	13	5/0	2/0	
d. RHR Complex				
1. Division I Pump Room	50	4/0		
2. Division II Pump Room	51	4/0		
3. EDG 11 Room Suppression				0/8
4. EDG 12 Room Suppression				0/8
5. EDG 13 Room Suppression				0/8
6. EDG 14 Room Suppression				0/8
7. EDG 11 Switchgear Room	52	6/0		
8. EDG 12 Switchgear Room	53	6/0		
9. EDG 13 Switchgear Room	54	6/0		
10. EDG 14 Switchgear Room	55	6/0		
e. General Service Water Pump House				
1. First Floor	31	2/0		3/0

*(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 and notification.) instruments.

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

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.10 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 a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.7.10 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

3.3.7.11 The radioactive liquid 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 3.11.1.1 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 3.3.7.11-1. Restore the inoperable instrumentation to OPERABLE within the time specified in the ACTION or, pursuant to Specification 6.9.1.8, explain in the next Semiannual Radioactive Effluent Release Report why the 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

4.3.7.11 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 4.3.7.11-1.

TABLE 3.3.7.11-1

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 D11-N007	1	110
2. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Circulating Water Reservoir Decant Line D11-N402	1	111
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line G11R037	1	112
b. Circulating Water Reservoir Decant Line N71-R802	1	112

TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

ACTION 110 - 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 individuals independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 111 - 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 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/mL, for 137-Cs.

ACTION 112 - 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 may be used to estimate flow.

TABLE 4.3.7.11-1

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	P	P	R(3)	Q(1)(2)
2. GROSS BETA OR GAMMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Circulating Water Reservoir Decant Line D11-N402	D	M	R(3)	Q(5)
3. FLOW RATE MEASUREMENT DEVICES (4)				
a. Liquid Radwaste Effluent Line	D(4)	N.A.	R	Q
b. Circulating Water Reservoir Decant Line	D(4)	N.A.	R	Q

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TABLE 4.3.7.11-1 (Continued)

TABLE NOTATIONS

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

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

2. Circuit failure.

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

1. Instrument indicates measured levels above the alarm setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(3) The initial calibration shall be done using vendor supplied calibration data and a qualitative source check. Subsequent channel calibration performed during plant operations shall be based upon comparisons with appropriate system grab samples.

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

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

1) Instrument indicates measured levels above the alarm setpoint.

2) Circuit failure.

3) Instrument indicates a downscale failure.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.7.12-1 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 3.3.7.12-1

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 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or, pursuant to Specification 6.9.1.8, explain in the next Semiannual Radioactive Effluent Release Report why the 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

4.3.7.12 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 4.3.7.12-1.

TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. REACTOR BUILDING EXHAUST PLENUM EFFLUENT MONITORING SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm	1	*	121
b. Iodine Sampler	1	*	122
c. Particulate Sampler	1	*	122
d. Sampler Flow Rate Monitor	1	*	123
2. OFFGAS MONITORING SYSTEM (At the 2.2 minute delay piping)			
a. Hydrogen Monitor	1	**	124
b. Noble Gas Activity Monitor	1	***	126
3. STANDBY GAS TREATMENT SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm	1	*	125
b. Iodine Sampler	1	*	122
c. Particulate Sampler	1	*	122
d. Effluent System Flow Rate Monitor	1	*	123
e. Sampler Flow Rate Monitor	1	*	123

TABLE 3.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. TURBINE BLDG. VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1	*	121
b. Iodine Sampler	1	*	122
c. Particulate Sampler	1	*	122
d. Sampler Flow Rate Monitor	1	*	123
5. SERVICE BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor	1	*	121
b. Iodine Sampler	1	*	122
c. Particulate Sampler	1	*	122
d. Sampler Flow Rate Monitor	1	*	123

TABLE 3.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
6.	RADWASTE BUILDING VENTILATION MONITORING SYSTEM			
	a. Noble Gas Activity Monitor	1	*	121
	b. Iodine Sampler	1	*	122
	c. Particulate Sampler	1	*	122
	d. Sampler Flow Rate Monitor	1	*	123
7.	ONSITE STORAGE BUILDING VENTILATION EXHAUST RADIATION MONITOR			
	a. Noble Gas Activity Monitor	1	*	121
	b. Iodine Sampler	1	*	122
	c. Particulate Sampler	1	*	122
	d. Sampler Flow Rate Monitor	1	*	123

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATIONS

* At all times.

** During main condenser offgas treatment system operation.

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

ACTION STATEMENTS

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 grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 122 - With the number of channnels OPERABLE one 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.

ACTION 123 - 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 124 - 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 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 125 - 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 4 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 126 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the holdup system may be released to the environment provided that prior to initiating the release:

a. The offgas system is not bypassed, and

b. The offgas delay system noble gas activity effluent (downstream) monitor is OPERABLE;

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

TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. REACTOR BUILDING EXHAUST PLENUM					
a. Noble Gas Activity Monitor - Providing Alarm	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
2. OFFGAS MONITORING SYSTEM (At the 2.2 minute delay piping)					
a. Hydrogen Monitor	D	N.A.	Q(3)	M	**
b. Noble Gas Activity Monitor	D	M	R(3)	Q(1)	***
3. STANDBY GAS TREATMENT MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. TURBINE BLDG. VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(4)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
5. SERVICE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(4)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. RADWASTE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(4)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
7. ONSITE STORAGE BUILDING VENTILATION EXHAUST RADIATION MONITOR					
a. Noble Gas Activity Monitor	D	M	R(2)	Q(4)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3.7.12-1 (Continued)

TABLE NOTATIONS

* At all times.

** During main condenser offgas treatment system operation.

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

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

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

(2) The initial CHANNEL CALIBRATION shall be done using vendor supplied calibration data and a qualitative source check. Subsequent channel calibration performed during plant operations shall be based upon comparisons with appropriate system grab samples.

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

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

(4) The CHANNEL CALIBRATION shall also demonstrate that automatic isolation occurs on high level and that control room alarm annunciation occurs if any of the following conditions exists:

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

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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 low pressure stop valve or intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours or close at least one valve in the affected steam lead(s) 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 14 days by cycling each of the following valves through at least one complete cycle from the running position and observing valve closure:

   1. Four high pressure turbine stop valves,

   2. Six low pressure turbine low pressure stop valves,

   3. Four high pressure turbine control valves, and

   4. Six low pressure turbine intercept valves.

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

c. At least once per 40 months by dismantling and inspecting 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 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system 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 feedwater/main turbine trip 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 either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

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

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

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each feedwater/main turbine trip 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.

TABLE 3.3.9-1

FEEDWATER/MAIN TURBINE SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNIT	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM	APPLICABLE OPERATIONAL CONDITIONS
a. Reactor Vessel High Water Level - Level 8	2	1

TABLE 3.3.9-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel High Water Level - Level 8	$\leq$ 214 inches*	$\leq$ 219 inches

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

TABLE 4.3.9.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Reactor Vessel High Water Level - Level 8	S	M	R	1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, immediately 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 measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 105% and 102.5%, respectively, of rated core flow, at least once per 18 months.

---

*See Special Test Exception 3.10.4.

**If not performed within the previous 31 days.

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours* by 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 the recirculation pumps are operating at the same speed:

a. The indicated recirculation loop flow differs by more than 10% from the established pump speed-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 the mean of all jet pump differential pressures in the same loop by more than 10%.

*During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

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RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation pump speed shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

a. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

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 145°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.

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3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

5 safety/relief valves @ 1110 psig $\pm$1%
5 safety/relief valves @ 1120 psig $\pm$1%
5 safety/relief valves @ 1130 psig $\pm$1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 95°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 95°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve position indicators inoperable, restore the inoperable indicator(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.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The valve position indicator for each safety/relief valve shall be demonstrated OPERABLE with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a:

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

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

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

**Any portion of this surveillance requirement which requires entry into the primary containment and whose surveillance interval expires when the primary containment is inerted may be rescheduled to the next time the primary containment is not inerted.

***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) Open	Close	Low-Low Set Function Allowable Value (psig) Open	Close
F013A (V22-2071)	1017	905	1037	*
F013G (V22-2068)	1047	935	1067	*

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 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 both 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.

SURVEILLANCE REQUIREMENTS

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

a. CHANNEL FUNCTIONAL TEST 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.

*Closing pressure must be at least 100 psi less than actual opening pressure.

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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 primary containment atmosphere gaseous radioactivity monitoring system channel.

b. The primary containment sump flow monitoring system consisting of:

   1. The drywell floor drain sump level, flow and pump-run-time system, and

   2. The drywell equipment drain sump level, flow and pump-run-time system.

c. The drywell floor drain level monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Primary containment atmosphere 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. Primary containment sump flow and drywell floor drain level 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.

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REACTOR COOLANT SYSTEM

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 1040 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

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

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valve, 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 high/low pressure 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 setpoint 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 in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which has 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.

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 primary containment atmospheric gaseous radioactivity at least once per 4 hours,*

b. Monitoring the primary containment sump flow rate at least once per 4 hours,

c. Monitoring the drywell floor drain sump level at least once per 4 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 high/low pressure 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.

*Not a means of quantifying leakage.

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TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	VALVE DESCRIPTION
1. RHR System	
E11-F015A (V8-2161)	LPCI Loop A Injection Isolation Valve
E11-F015B (V8-2162)	LPCI Loop B Injection Isolation Valve
E11-F050A (V8-2163)	LPCI Loop A Injection Line Testable Check Valve
E11-F050B (V8-2164)	LPCI Loop B Injection Line Testable Check Valve
E11-F023 (V8-2171)	RPV Head Spray Outboard Isolation Valve
E11-F022 (V8-2172)	RPV Head Spray Inboard Isolation Valve
E11-F008 (V8-2092)	Shutdown Cooling RPV Suction Outboard Isolation Valve
E11-F009 (V8-2091)	Shutdown Cooling RPV Suction Inboard Isolation Valve
E11-F608 (V8-3407)	Shutdown Cooling Suction Isolation Valve
2. Core Spray System	
E21-F005A (V8-2021)	Loop A Inboard Isolation Valve
E21-F005B (V8-2022)	Loop B Inboard Isolation Valve
E21-F006A (V8-2023)	Loop A Containment Check Valve
E21-F006B (V8-2024)	Loop B Containment Check Valve
3. High Pressure Coolant Injection System	
E41-F007 (V8-2193)	Pump Discharge Outboard Isolation Valve
E41-F006 (V8-2194)	Pump Discharge Inboard Isolation Valve
4. Reactor Core Isolation Cooling System	
E51-F012 (V8-2227)	Pump Discharge Isolation Valve
E51-F013 (V8-2228)	Pump Discharge to Feedwater Header Isolation Valve

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E11-F015A & B, E11-F022, F023, E11-F050A & B	RHR LPCI	482 ± 12
E11-F008, F009, F608	RHR Shutdown Cooling	138 ± 3
E21-F005A & B, E21-F006A & B	Core Spray	440 ± 12
E41-F006, F007	HPCI	70 ± 1
E51-F013, F014	RCIC	70 ± 1

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REACTOR COOLANT SYSTEM

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 limit 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

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

   2. The provisions of Specification 3.0.3 are not applicable.

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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.

c. Continuously recording the conductivity of the reactor coolan or, when the continuous recording conductivity monitor is inoperab. by 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.

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

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

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 microcurie 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 CONDITION 1, 2, or 3 with the specific activity of the primary coolant;

1. Greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 but less than or equal to 4 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 microcurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 microcurie 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 microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

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

b. In OPERATIONAL CONDITION 1, 2, 3, or 4, with the specific activity of the primary coolant greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4.a) 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 pursuant to Specification 6.9.2 within 30 days. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcurie per gram DOSE EQUIVALENT I-131 together with the following additional information.

ACTION: (Continued)

c. In OPERATIONAL CONDITION 1 or 2, with:

1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour*, or
2. The off-gas level, at the delay pipe, increased by more than 10,000 microcuries per second in 1 hour during steady-state operation at release rates less than 75,000 microcuries per second, or
3. The off-gas level, at the delay pipe, 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. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.2 at least once per 92 days containing the results of the specific activity analysis together with the below additional information for each occurrence.

Additional Information

1. Reactor power history starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.
2. Fuel burnup by core region.
3. Clean-up flow history starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.
4. Off-gas level starting 48 hours prior to:
   a) The first sample in which the limit was exceeded, and/or
   b) The THERMAL POWER or off-gas level change.

SURVEILLANCE REQUIREMENTS

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

*Not applicable during the startup test program.

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 $\bar{E}$ 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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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 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-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 71°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; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined 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, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

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 Part 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 71°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.

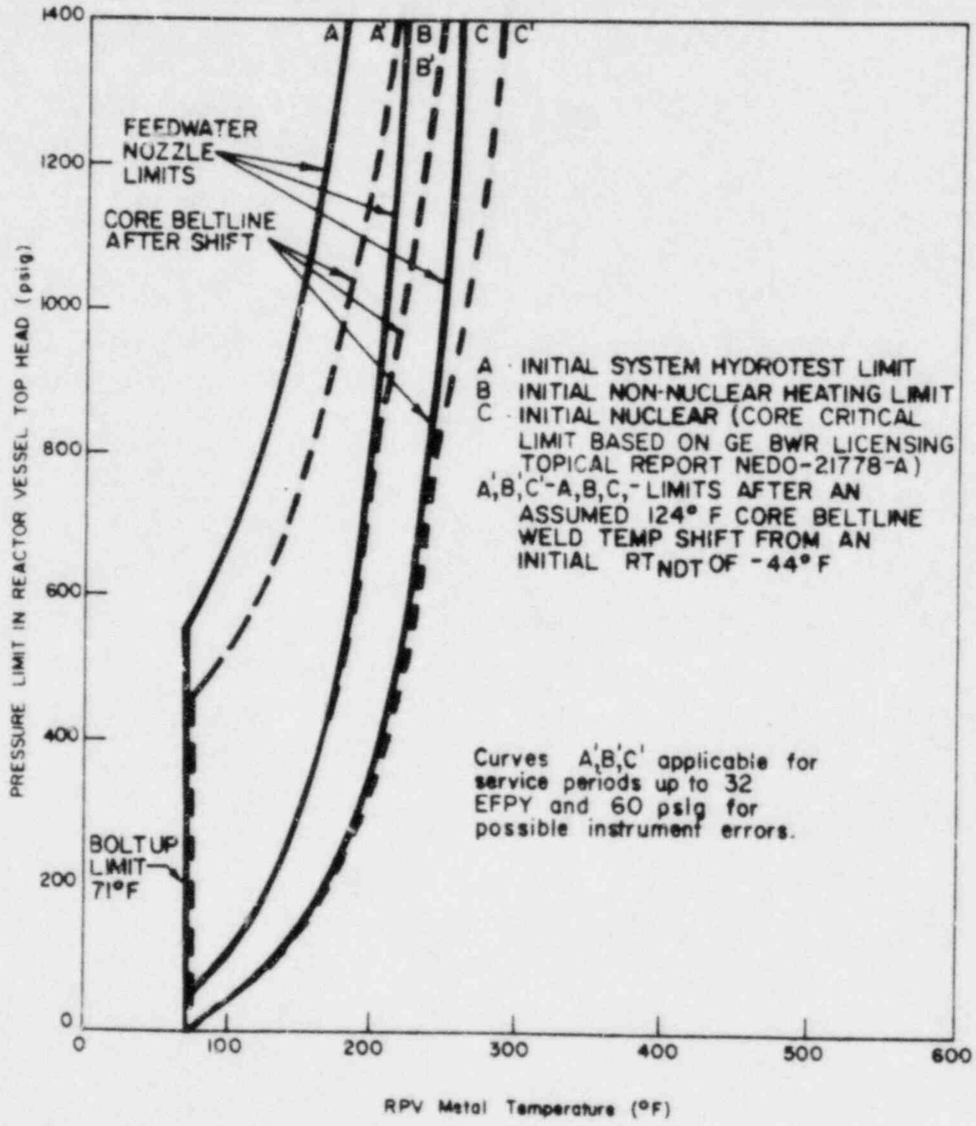

FIGURE 3.4.6.1-1

MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL 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	WITHDRAWAL TIME (EFPY)
1	Azimuth 300	0.7	8
2	Azimuth 120	0.7	24
3	Azimuth 30	0.7	Standby

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1101 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1101 psig, reduce the pressure to less than 1101 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1101 psig at least once per 12 hours.

*Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 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 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

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3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the reactor coolant system temperature above 200°F.

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8 No requirements other than Specification 4.0.5 and the guidelines of NUREG-0313, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Revision 1.

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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. One OPERABLE RHR heat exchanger.

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 1 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 in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

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

#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. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 1 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 in operation, within 1 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 one shutdown cooling mode loop of the residual heat removal system or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#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 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 The emergency core cooling systems shall be OPERABLE with:

a. The core spray system (CSS) consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE CSS pumps, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

b. The low pressure coolant injection (LPCI) system of the residual heat removal system consisting of two subsystems with each subsystem comprised of:

   1. Two OPERABLE LPCI (RHR) pumps, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

c. The high pressure cooling injection (HPCI) system consisting of:

   1. One OPERABLE HPCI pump, and

   2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

d. The automatic depressurization system (ADS) with at least five OPERABLE ADS valves.

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

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*The HPCI system is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

**The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 150 psig.

#See Special Test Exception 3.10.6.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the core spray system:

1. With one CSS subsystem inoperable, provided that at least one LPCI pump in each LPCI subsystem is OPERABLE, restore the inoperable CSS 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.

2. With both CSS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. For the LPCI system:

1. With one LPCI pump in either or both LPCI subsystems inoperable, provided that at least one CSS subsystem is OPERABLE, restore the inoperable LPCI pump(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.

2. With a LPCI system cross-tie valve closed, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With one LPCI subsystem otherwise inoperable, provided that both CSS subsystems are OPERABLE, restore the inoperable LPCI 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.

4. With both LPCI subsystems otherwise inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.*

c. For the HPCI system, provided the CSS, the LPCI system, the ADS and the RCIC system are OPERABLE:

1. With the HPCI system inoperable, restore the HPCI 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 $\leq$ 150 psig 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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LIMTING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the CSS and the LPCI system are OPERABLE, 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 $\leq$ 150 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 $\leq$ 150 psig within the next 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the CSS, the LPCI system, and the HPCI system:

a) 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.

b) 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.

2. For the LPCI system, verifying that the cross-tie valve is open.

3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

---

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

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b. Verifying that, when pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure of greater than or equal to ( ) psig, corresponding to a reactor vessel pressure of $\geq$ 115 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure of $\geq$ ( ) psig, corresponding to a reactor vessel to primary containment differential pressure of $\geq$ 20 psig.

3. The HPCI pump develops a flow of at least 5000 gpm against a test line pressure of $\geq$ 1100 psig when steam is being supplied to the turbine at 1000 +20 , -80 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, 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 exluded from this test.

2. For the HPCI system, verifying that:

   a) The system develops a flow of at least 5000 gpm against a test line pressure of 265 psig, corresponding to a reactor vessel pressure of $\geq$ 165 psig, when steam is being supplied to the turbine at 165 + 50, - 0 psig.*

   b) The suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - 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 test.

SURVEILLANCE REQUIREMENTS (Continued)

d. For the ADS:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the primary containment pneumatic supply system low pressure alarm system.

2. At least once per 18 months:

   a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 150 psig* and observing that either:

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

      2) There is a corresponding change in the measured steam flow.

   c) Performing a CHANNEL CALIBRATION of the primary containment pneumatic supply system low pressure alarm system and verifying an alarm setpoint of 80 ± 5 psi on decreasing pressure.

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*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.

3/4 5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.2 At least two of the following subsystems shall be OPERABLE:

a. Core spray system (CSS) subsystems with a subsystem comprised of:

1. At least two OPERABLE CSS pumps, and

2. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

a) From the suppression chamber, or

b) When the suppression chamber water level is less than the limit required in Specification 3.5.3 or is drained, from the condensate storage tank containing at least 150,000 available gallons of water, equivalent to a level of 18 feet.

b. Low pressure coolant injection (LPCI) system subsystems with a subsystem comprised of:

1. At least two OPERABLE LPCI (RHR) pumps, and

2. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel**.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

a. With one of the above required subsystem(s) inoperable, restore at least two subsystem(s) to OPERABLE status within 4 hours or suspend all operations with a potential for draining the reactor vessel.

b. With both of the above required subsystems inoperable, suspend CORE ALTERATIONS and all operations with a potential for draining the reactor vessel. Restore at least one subsystem to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY 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 spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

**Upon receipt of a LPCI initiation signal, operator action is required to manually open the torus suction valves to facilitate LPCI operation if the LPCI system is in the RHR shutdown cooling mode of operation per Specification 3.4.9.2.

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 The core spray system shall be determine OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.a.2.b).

3/4.5.3 SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3 with a contained water volume of at least 121,080 ft^3, equivalent to a level of 14'4" (-2 inches indication).

b. In OPERATIONAL CONDITIONS 4 and 5* with a contained volume of at least 64,550 ft^3, equivalent to a level of 9'0" (-66 inches indication), except that the suppression chamber 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 150,000 available gallons of water, equivalent to a level of 18', and

4. The core spray system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

---

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

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

a. 14'4" (-2 inch indication) at least once per 24 hours in OPERATIONAL CONDITIONS 1, 2, and 3.

b. 9'0" (-66 inch indication) at least once per 12 hours in OPERATIONAL CONDITIONS 4 and 5*.

4.5.3.2 With the suppression chamber 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 chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, 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 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of 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 P_a, 56.5 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blank flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

*See Special Test Exception 3.10.1.

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

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to: L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 56.5 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests when pressurized to P_a, 56.5 psig.

c. *Less than or equal to 100 scf per hour for all four main steam lines when tested at 25.0 psig.

d. A combined leakage rate of less than or equal to 5 gpm for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 62.2 psig.

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

ACTION:

With:

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

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines, or

d. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 5 gpm,

Prior to increasing reactor coolant system temperature above 200°F restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* and valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

*Exemption to Appendix J of 10 CFR Part 50.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. The leakage rate to less than or equal to 100 scf per hour for all four main steam lines, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which pentrate the primary containment to less than or equal to 5 gpm.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined using the methods and provisions described herein:

a. Integrated Primary Containment Leakage Rate - Type A Test

1. Integrated leak rate tests shall be performed at the test pressure (P_a) of 56.5 psig. Containment pressure shall not be permitted to decrease more than 1 psi below P_a.

2. Type B and C tests should be completed prior to each Type A test. Type B and C leakages not accounted for in the Type A test shall be added to the upper confidence limit (UCL) to estimate the overall integrated leakage rate. However, when adding the leakage rate measured during a Type C test to the results of a Type A test, the lower leakage rate of the two isolation valves in a line shall be used.

3. If the leakage rate exceeds the acceptance criterion, corrective action shall be required. If, during the performance of a Type A test, excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with the satisfactory completion of the test, these leakage paths may be isolated and the Type A test continued until completion. A local leakage test shall be performed before and after the repair of each isolated leakage path. The sum of the post repaired local leakage rates and the UCL shall be less than 75% of the maximum allowable leakage rate, L_a.

4. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

5. A Type A test shall last a minimum of 8 hours after stabilization. The following criteria shall be met in order to consider a Type A test satisfactorily completed:

SURVEILLANCE REQUIREMENTS (Continued)

(a) After attaining test pressure, the containment atmosphere shall be allowed to stabilize for a period of about 4 hours prior to the start of the Type A test. The atmosphere shall be considered stabilized when the rate of change of average weighted temperature is less than 1.0°F/hour averaged over the last 2 hours, or the rate of change of temperature changes less than 0.5°F/hour averaged over the last 2 hours.

(b) The magnitude of the calculated leak rate, based on mass plot calculations, is tending to stabilize at a value less than the maximum allowable operational leak rate of 0.75 L_a.

(c) The end of test upper 95% confidence limit for the calculated leak rate based on mass plot calculations shall be less than the maximum allowable operational leak rate of 0.75 L_a.

(d) The mean of the measured leak rates based on mass plot calculations over the last 5 hours of test or last 20 data points, whichever provides the most data, shall be less than the maximum allowable operation leak rate of 0.75 L_a.

(e) Data shall be recorded at approximately equal intervals and in no case at intervals greater than 1 hour.

(f) At least 20 data points shall be provided for proper statistical analysis.

6. Test and Analysis Method

(a) The absolute test method shall be used.

(b) The mass plot (mass point) analysis technique, as described in ANSI/ANS-56.8-1981, in addition to the method described in BN-TOP-1, shall be used to compute the containment leakage rate.

(c) An upper one sided 95% confidence limit for the leakage rate shall be determined based upon normal regression theory.

7. Instrumentation

(a) During Type A testing, measurements shall be made of containment dry bulb temperature, dewpoint temperature, and absolute pressure.

(b) Dry bulb temperature sensors shall have an accuracy of ±0.5°F or better over the temperature range expected during the test ±20°F and a repeatability of at least 0.1°F.

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SURVEILLANCE REQUIREMENTS (Continued)

(c) Dewpoint temperature sensors shall have an accuracy of ±1°F or better over the dewpoint temperature range expected during the test ±20°F and a repeatability of at least ±0.5°F.

(d) Pressure sensors should have a range such that P_a is between 25 and 75% of full scale. Accuracy shall be at least 0.015% of full scale with resolution and repeatability of 0.001% of full scale.

(e) The number and location of temperature and dewpoint sensors shall be determined prior to each Type A test based on a temperature survey of the containment.

(f) At least two-thirds of the dry bulb temperature sensors must be functioning properly during the test.

(g) At least two-thirds of the dewpoint temperature sensors shall be functioning properly during the test. However, if data recorded over the last 5 hours indicate that dewpoint temperatures have stabilized and any changes are not of an order to cause error in leak rate calculations, then malfunction of any or all of the dewpoint sensors shall not require aborting the test.

(h) At least one precision pressure gage shall be functioning properly during the test.

(i) Prior to each Type A test and following the failure of any sensor, an instrument error analysis shall be performed using the Instrument Selection Guide (ISG) formula of ANSI/ANS-56.8-1981. The ISG shall not exceed 0.25 L_a at the end of a test except as noted in (g) above.

8. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 56.5 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.

9. If any periodic Type A test fails to meet 0.75 L_a, 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 L_a, a type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

10. The accuracy of each Type A test shall be verified by a supplemental test which:

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SURVEILLANCE REQUIREMENTS (Continued)

(a) Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.

(b) Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

(c) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at P_a, 56.5 psig,

b. Type B and C tests shall be conducted with gas at P_a, 56.5 psig*, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves,

3. Penetrations using continuous leakage monitoring systems,

4. Valves pressurized with fluid from a seal system,

5. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

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

c. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3.

d. Main steam line isolation valves shall be leak tested at least once per 18 months.

e. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 56.5 psig, at intervals no greater than once per 3 years.

f. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 62.2 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

*Unless a hydrostatic test is required per Table 3.6.3-1.

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SURVEILLANCE REQUIREMENTS (Continued)

g. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

h. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Specification 4.6.1.8.2.

i. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2a., 4.6.1.2b. and 4.6.1.2c.

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PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 56.5 psig.

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

ACTION:

a. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

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

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the primary containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

---

*See Special Test Exception 3.10.1.

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to Pa, 56.5 psig.

b. By conducting an overall air lock leakage test at P_a, 56.5 psig, and by verifying that the overall air lock leakage rate is within its limit:

1. After each opening, unless performed within the previous 6 months,* but at least once per 18 months,* and

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the airlock sealing capability.**

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

*The provisions of Specification 4.0.2 are not applicable.

**Exemption to Appendix J of 10 CFR 50.

***Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been de-inerted.

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE with each subsystem comprised of a flow path from the associated control air division to the main steam lines.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by cycling each testable valve except the motor-operated MSIVs through at least one complete cycle of full travel.

b. During each COLD SHUTDOWN, if not performed within the previous 31 days, by cycling each valve including the motor-operated MSIVs not testable during operation through at least one complete cycle of full travel.

c. At least once per 18 months by performance of a functional test of the subsystem throughout its operating sequence.

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PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the structure, the inspection procedure, the inspection criteria, and the corrective actions taken.

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -0.50 and +2.00 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure 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.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits at least once per 12 hours.

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DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric 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 (At least one at each elevation)
a.	574'1"	160°, 200°, 300° or 330°
b.	597'0"	35°, 75°, 93°, 135°, 175°, 200°, 246°, 272°, 306° or 345°
c.	621'8"	0°, 90°, 180° or 270°
d.	648'6"	45°, 135°, 225° or 315°
e.	662'0"	0°, 90°, 180° or 270°
f.	665'6"	0° or 180°

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DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system (6-inch, 10-inch, 20-inch, and 24-inch valves) may be in operation with the supply and exhaust isolation valves in one supply line and one exhaust line open for inerting, deinerting or pressure control.* Purge/vent operations through the SGTS shall be limited to 90 hours each 365 days.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With a drywell and suppression chamber purge system supply and/or exhaust isolation valve open, except as permitted above, close the 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.

b. With a drywell and suppression chamber purge system supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Specification 4.6.1.8.2, 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

4.6.1.8.1 Before being opened for purge/vent operation through SGTS, the drywell and suppression chamber purge supply and exhaust butterfly isolation valves shall be verified not to have been open for purge/vent operation through SGTS for more than 90 hours in the previous 365 days.*

4.6.1.8.2 At least once per 92 days each penetration for each 6-inch, each 10-inch, each 20-inch, and each 24-inch drywell and suppression chamber purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

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*Valves open for pressure control are not subject to the 90 hour per 365 day limit provided the 6-inch bypass line is being utilized.

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3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 121,080 ft^3 and 124,220 ft^3, equivalent to a level between 14'4" (-2 inches indication) and 14'8" (+2 inches indication) and a

2. Maximum average temperature of 95°F except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

c) 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 1 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber 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 chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the 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.

2. With the suppression chamber 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 chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool water temperature instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify suppression chamber water temperature to be within the limits at least once per 12 hours.

d. With more than one suppression pool water temperature instrumentation channel inoperable, restore at least seven temperature instrumentation channels 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.

e. With the drywell-to-suppression chamber leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber 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 chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

   2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:

      a) Suppression chamber average water temperature to be less than or equal to 110°F, and

      b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.

   3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

SURVEILLANCE REQUIREMENTS (Continued)

c. By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 160°F and reactor coolant system pressure greater than 200 psig.

d. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

e. By verifying eight suppression pool water temperature instrumentation channels OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

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

3. CHANNEL CALIBRATION at least once per 18 months, with the water high temperature alarm setpoint for ≤ 105°F.

f. At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 1 psi and verifying that the differential pressure does not decrease by more than 0.20 inch of water per minute for a period of 10 minutes. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

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SUPPRESSION POOL AND DRYWELL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool and drywell spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool and drywell spray spargers.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool and/or drywell spray loop inoperable, restore the inoperable loop 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. With both suppression pool and/or drywell spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool and drywell spray mode of the RHR system shall be demonstrated OPERABLE:

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

b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and suppression pool spray sparger when tested pursuant to Specification 4.0.5.

c. By performance of an air or smoke flow test of the drywell spray nozzles at least once per 5 years and verifying that each spray nozzle is unobstructed.

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*Whenever both 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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SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

a. One OPERABLE RHR pump, and

b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

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

b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

*Whenever both 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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3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

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

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blank flange.*

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

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either:

   1. The inoperable valve is returned to OPERABLE status, or

   2. The instrument line is isolated and the associated instrument is declared inoperable.

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

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

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SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
A. Automatic Isolation Valves[a]	
1. Group 1 - Main Steam System	
Main Steam Isolation Valves (MSIVs)	
Inboard	
Line A: B21-F022A (V17-2003)	5
Line B: B21-F022B (V17-2001)	5
Line C: B21-F022C (V17-2002)	5
Line D: B21-F022D (V17-2004)	5
Outboard	
Line A: B21-F028A (V17-2007)	5
Line B: B21-F028B (V17-2005)	5
Line C: B21-F028C (V17-2006)	5
Line D: B21-F028D (V17-2008)	5
Main Steam Line Drains Isolation Valves	
Inboard: B21-F016 (V17-2009)	23
Outboard: B21-F019 (V17-2010)	23
2. Group 2 - Reactor Water Sample System	
Reactor Water Sample Line Isolation Valves	
Inboard: B31-F019 (V17-2077)	15
Outboard: B31-F020 (V17-2078)	15

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
A. Automatic Isolation Valves[a] (Continued)	
3. Group 3 - Residual Heat Removal (RHR) System	
RHR Drywell Spray Isolation Valves	
Loop A: E11-F016A (V8-2167)	150
E11-F021A (V8-2169)	60
Loop B: E11-F016B (V8-2168)	150
E11-F021B (V8-2170)	60
RHR Containment Cooling/Test Isolation Valves[b]	
Loop A: E11-F024A (V8-2135)	60
Loop B: E11-F024B (V8-2136)	60
RHR Suppression Pool Spray Isolation Valves	
Loop A: E11-F027A (V8-2157)	60
Loop B: E11-F027B (V8-2158)	60
RHR Suppression Pool Spray/Test Isolation Valves	
Loop A: E11-F028A (V8-2155)	60
Loop B: E11-F028B (V8-2156)	60
4. Group 4 - Residual Heat Removal Shutdown Cooling and Head Spray	
RHR Shutdown Cooling Suction Isolation Valves	
Inboard: E11-F009 (V8-2091)	52
Outboard: E11-F008 (V8-2092)	52
RHR Reactor Pressure Vessel Head Spray Isolation Valves	
Inboard: E11-F022 (V8-2172)	36
Outboard: E11-F023 (V8-2171)	120

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
A. Automatic Isolation Valves[(a)] (Continued)	
5. Group 5 - Core Spray System	
Core Spray Pump Flow Test Valves[(b)]	
Loop A: E21-F015A (V8-2033)	150
Loop B: E21-F015B (V8-2034)	150
6. Group 6 - High Pressure Coolant Injection (HPCI) System	
HPCI Turbine Steam Supply Isolation Valves	
Inboard: E41-F002 (V17-2020)	15
Outboard: E41-F003 (V17-2021)	15
HPCI Turbine Steam Supply Outboard Isolation Bypass Valve	
E41-F600 (V17-2088)	15
HPCI Booster Pump Suction from Suppression Chamber Isolation Valve	
E41-F042 (V8-2202)[(b)]	60
7. Group 7 - High Pressure Coolant Injection (HPCI) Vacuum Breakers	
HPCI Turbine Exhaust Line Vacuum Breaker Isolation Valves	
E41-F075 (V11-2013)	60
E41-F079 (V11-2019)	60

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
A. Automatic Isolation Valves[a] (Continued)	
8. Group 8 - Reactor Core Isolation Cooling (RCIC) System	
RCIC Steam Line Isolation Valves	
Inboard: E51-F007 (V17-2030)	15
Outboard: E51-F008 (V17-2031)	15
9. Group 9 - Reactor Core Isolation Cooling (RCIC) System Vacuum Breakers	
RCIC Turbine Exhaust Line Vacuum Breaker Isolation Valves	
E51-F062 (V11-2020)	60
E51-F084 (V11-2026)	60
10. Group 10 - Reactor Water Cleanup (RWCU) System (Inboard)	
Inboard: G33-F001 (V8-2252)	10
11. Group 11 - Reactor Water Cleanup (RWCU) System (Outboard)[c]	
Outboard: G33-F004 (V8-2253)	10
12. Group 12 - Torus Water Management System (TWMS)	
TWMS to RHR Line Isolation Valves[b][d]	
G51-F605 (V8-3847)	60
G51-F604 (V8-3849)	60
TWMS to CSS Test Line Isolation Valves[b][d]	
G51-F607 (V8-3848)	60
G51-F606 (V8-3850)	60
Torus Drain Isolation Valves[b][d]	
G51-F600 (V8-3832)	60
G51-F602 (V8-3831)	60
G51-F601 (V8-3834)	60
G51-F603 (V8-3833)	60

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
A. Automatic Isolation Valves[a] (Continued)	
13. Group 13 - Drywell Sumps	
Drywell Floor Drain Sump Pump Discharge Isolation Valves	
G11-F600 (V9-2044)	15
G11-F003 (V9-2005)	15
Drywell Equipment Drain Sump Pump Discharge Isolation Valves	
G11-F018 (V9-2022)	15
G11-F019 (V9-2023)	15
14. Group 14 - Drywell and Suppression Pool Ventilation System	
Drywell Exhaust Isolation Valves	
T48-03-F602 (VR3-3024)	5
T46-F411 (VR3-3026)	5
T46-F402 (VR3-3023)	5
Drywell N_2 and Air Purge Inlet Isolation Valves	
T48-03-F601 (VR3-3011)	5
T48-F408 (V4-2060)	5
T48-F407 (VR3-3012)	5
Suppression Pool Exhaust Air Purge to Standby Gas Treatment System and N_2 Inlet Isolation Valves	
T46-F400 (VR3-3015)	5
T48-F410 (V4-2063)	5
T46-F401 (VR3-3016)	5
T46-F412 (VR3-3019)	5

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
A. Automatic Isolation Valves[(a)] (Continued)	
14. Group 14 - Drywell and Suppression Pool Ventilation System (Continued)	
Suppression Pool N_2 and Air Purge Inlet Isolation Valves	
T48-F404 (VR3-3013)	5
T48-F405 (VR3-3014)	5
T48-F409 (V4-2061)	5
15. Group 15 - Traversing In-core Probe (TIP) System	
Tip System Ball Valves A, B, C, D and E	NA
16. Group 16 - Nitrogen Inerting System	
N_2 Pressure Control Isolation Valves	
Inboard: T48-F455 (VR3-2825)	60
Outboard: T48-F453 (VR3-2823)	60
T48-F454 (VR3-2824)	60
T48-F456 (VR3-2826)	60
T48-F457 (VR3-2827)	60
T48-F458 (VR3-2828)	
17. Group 17 - Recirculation Pump System	
Recirculation Pumps Seal Purge Isolation Valves	
Inboard: B31-F014A (V8-3710)	5
B31-F014B (V8-3590)	5
Outboard: B31-F016A (V8-3767)	5
B31-F016B (V8-3768)	5
18. Group 18 - Primary Containment Pneumatic Supply System	
N_2 to Drywell Isolation Valves	
Inboard: T49-F601 (V4-2080)	60
T49-F602 (V4-2188)	60
Outboard: T49-F465 (V4-2079)	60
T49-F468 (V4-2187)	60

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves[(e)]	
1. Main Steam Isolation Valves (MSIV) Leakage Control Valves	NA
B21-F434 (V5-2294)	
2. RHR Shutdown Cooling Suction Inboard Isolation Valve Bypass Valve[(q)]	NA
E11-F608 (V8-3407)	
3. LPCI Inboard Isolation Valves[(f)]	NA
Loop A: E11-F015A (V8-2161)	
E11-F406 (V13-7687)	
Loop B: E11-F015B (V8-2162)	
E11-F407 (V13-7688)	
4. RHR Pumps Recirculation Motor Operated Valves[(b)(g)]	NA
Pumps A/C: E11-F007A (V8-2133)	
Pumps B/D: E11-F007B (V8-2134)	
5. Warmup and Flush Line Isolation Valve	NA
E11-F026 (V8-2152)	
6. Reactor Protection System Instrumentation Isolation Valves	NA
Division I: E11-F412 (V5-2546)	
E11-F413 (V5-2547)	
Division II: E11-F414 (V5-2548)	
E11-F415 (V5-2549)	
7. RHR Pump Torus Suction Isolation Valves[(b)]	NA
Pump A: E11-F004A (V8-2099)	
Pump B: E11-F004B (V8-2102)	
Pump C: E11-F004C (V8-2101)	
Pump D: E11-F004D (V8-2100)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves[(e)] (Continued)	NA
8. Core Spray Loop Inboard Isolation Valves	
Loop A: E21-F005A (V8-2021) Loop B: E21-F005B (V8-2022)	
9. Core Spray Loop Minimum Recirculation Isolation Valves[(b)(h)]	NA
Loop A: E21-F031A (V8-2031) Loop B: E21-F031B (V8-2032)	
10. Core Spray Loop Suction from Suppression Chamber Valves[(b)]	NA
Loop A: E21-F036 A (V8-2007) Loop B: E21-F036 B (V8-2008)	
11. HPCI Pump Discharge to Reactor Feedwater Header Valve[(i)]	NA
E41-F006 (V8-2194)	
12. HPCI Pump Minimum Flow Valve[(b)(j)]	NA
E41-F012 (V8-2196)	
13. RCIC Pump Discharge to Feedwater Header Isolation Valve[(k)]	NA
E51-F013 (V8-2228)	
14. RCIC Pump Minimum Flow Valve[(b)(l)]	NA
E51-F019 (V8-2230)	

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves[e] (Continued)	
15. RCIC Pump Suction from Suppression Chamber Isolation Valves	NA
Inboard: E51-F031 (V8-2225)[b]	
16. Combustible Gas Control System Suction Isolation Valves	NA
Inboard	
Torus: Division I: T48-F602A (V4-2142)	
Division II: T48-F602B (V4-2141)	
Drywell: Division I: T48-F603A (V4-2144)	
Division II: T48-F603B (V4-2143)	
Outboard	
Torus: Division I: T48-F606A (V4-2156)	
Division II: T48-F606B (V4-2155)	
Drywell: Division I: T48-F605A (V4-2154)	
Division II: T48-F605B (V4-2153)	
17. Combustible Gas Control System Return Isolation Valves	NA
Inboard: Division I: T48-F601A (V4-2140)	
Division II: T48-F601B (V4-2139)	
Outboard: Division I: T48-F604A (V4-2148)	
Division II: T48-F604B (V4-2149)	
18. Primary Containment Monitoring System Torus Return Isolation Valves	NA
Division I: T50-F408A (V5-2158)	
Division II: T50-F408B (V5-2166)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER			MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves(e) (Continued)			
19. Primary Containment Monitoring System Torus Suction Isolation Valves			NA
Division I:	T50-F407A	(V5-2157)	
Division II:	T50-F407B	(V5-2165)	
20. Drywell Atmosphere Sample Isolation Valves			NA
Division I:	T50-F401A	(V5-2151)	
	T50-F402A	(V5-2152)	
	T50-F403A	(V5-2153)	
	T50-F404A	(V5-2154)	
	T50-F405A	(V5-2155)	
	T50-F406A	(V5-2156)	
Division II:	T50-F401B	(V5-2159)	
	T50-F402B	(V5-2160)	
	T50-F403B	(V5-2161)	
	T50-F404B	(V5-2162)	
	T50-F405B	(V5-2163)	
	T50-F406B	(V5-2164)	
21. Drywell to Suppression Chamber Vacuum Breakers N_2 Supply Isolation Valves			NA
	T48-F416	(V4-2036)	
	T48-F417	(V4-2065)	
	T48-F418	(V4-2075)	
	T48-F419	(V4-2077)	
	T48-F420	(V4-2082)	
	T48-F421	(V4-2084)	
	T48-F422	(V4-2086)	
	T48-F423	(V4-2088)	
	T48-F424	(V4-2090)	
	T48-F425	(V4-2092)	
	T48-F426	(V4-2094)	
	T48-F427	(V4-2096)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves[e] (Continued)	
22. Drywell Pressure Instrumentation Isolation Valves	NA
Division I: T50-F420A (V5-2230)	
Division II: T50-F420B (V5-2231)	
23. Suppression Pool Level Instrumentation Isolation Valves	NA
Division I: E41-F401 (V5-2551)[b]	
T50-F412A (V5-2555)[b]	
E41-F400 (V5-2550)	
Division II: E41-F403 (V5-2553)[b]	
T50-F412B (V5-2556)[b]	
E41-F402 (V5-2552)	
24. EECW Supply to Drywell Equipment Isolation Valves	NA
Division I: P44-F607A (V8-2485)	
Division II: P44-F606B (V8-2484)	
25. EECW Return from Drywell Equipment Isolation Valves	NA
Division I: P44-F606A (V8-2486)	
P44-F616 (V8-3890)	
Division II: P44-F607B (V8-2483)	
P44-F615 (V8-3889)	
26. Drywell Condensate Supply Outboard Isolation Valve[g]	NA
P11-F616 (V8-2790)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves(e) (Continued)	
27. Service Air to Drywell Isolation Valves(q)	NA
Inboard: P50-F604 (V5-2007)	
Outboard: P50-F603 (V5-2006)	
28. TIP System Shear Valves(m)(r)	NA
C51-J004A	
C51-J004B	
C51-J004C	
C51-J004D	
C51-J004E	
29. Post Accident Sampling Isolation Valves	NA
a. Drywell Atmosphere Sample Suction Valves	
Division I: P34-F404B (V13-7375)	
P34-F403B (V13-7365)	
Division II: P34-F403A (V13-7364)	
P34-F404A (V13-7374)	
b. Suppression Pool Atmosphere Sample Suction Valves	
Division I: P34-F405B (V13-7367)	
P34-F406B (V13-7377)	
Division II: P34-F405A (V13-7366)	
P34-F406A (V13-7376)	

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
B. Remote-Manual Isolation Valves(e) (Continued)	
29. Post Accident Sampling Isolation Valves (Continued)	NA
c. Gaseous Sample Return Valves	
P34-F408 (V13-7369)	
P34-F410 (V13-7379)	
d. Pressurized Reactor Coolant Sample Suction Valves	
P34-F401A (V13-7360)	
P34-F401B (V13-7361)	
e. Liquid Sample Return Valves	
P34-F407 (V13-7368)	
P34-F409 (V13-7378)	
C. Manual Isolation Valves	
1. Drywell Condensate Supply Header Inboard Isolation Valve(q)	NA
P11-F126 (V8-3120)	
2. Drywell Control Air and N_2 Outboard Isolation Bypass Valve(q)	NA
T49-F007 (V4-2172)	
3. N_2 to Drywell Outboard Isolation Bypass Valve(q)	NA
T49-F016 (V8-4140)	

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves	
1. Main Feedwater Reverse Flow Check Valves	NA
B21-F010A (V12-2008)	
B21-F010B (V12-2007)	
B21-F076A (V12-2002)(n)	
B21-F076B (V12-2001)(n)	
2. LPCI Injection Reverse Flow Check Valves	NA
E11-F050A (V8-2163)	
E11-F050B (V8-2164)	
3. RHR Heat Exchanger Relief Valves	NA
E11-F001A (V22-2643)	
E11-F001B (V22-2642)	
4. RHR Heat Exchanger Outlet Line Relief Valves(b)(p)	NA
E11-F025A (V22-2025)	
E11-F025B (V22-2041)	
. RHR Pump Suction From Recirc Piping Reverse Flow Check Valve	NA
E11-F408 (V8-3874)	
6. RHR Shutdown Cooling Suction Relief Valve(b)(p)	NA
E11-F029 (V22-2033)	
7. RHR Pump Torus Suction Relief Valves(b)(p)	NA
E11-F030A (V22-2034)	
E11-F030B (V22-2037)	
E11-F030C (V22-2036)	
E11-F030D (V22-2035)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
8. Core Spray Loop Containment Reverse Flow Check Valves	NA
E21-F006A (V8-2023)	
E21-F006B (V8-2024)	
9. Core Spray Loop Pump Suction Relief Valves[(b)(p)]	NA
E21-F032A (V22-2019)	
E21-F032B (V22-2004)	
10. Core Spray Loop Pump Discharge Pressure Relief Valves[(b)]	NA
E21-F011A (V22-2120)	
E21-F012A (V22-2016)	
E21-F011B (V22-2119)	
E21-F012B (V22-2017)	
11. Excess Flow Check Valves[(r)]	NA
a. Jet Pump Instrumentation	
B21-F513A (V13-2324)	
B21-F513B (V13-2325)	
B21-F513C (V13-2326)	
B21-F513D (V13-2327)	
B21-F514A (V13-2328)	
B21-F514B (V13-2329)	
B21-F514C (V13-2330)	
B21-F514D (V13-2331)	
B21-F515A (V13-2332)	
B21-F515B (V13-2333)	
B21-F515C (V13-2334)	

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
11. Excess Flow Check Valves (r) (Continued)	NA
a. Jet Pump Instrumentation (Continued)	
B21-F515D (V13-2335)	
B21-F515E (V13-2336)	
B21-F515F (V13-2337)	
B21-F515G (V13-2338)	
B21-F515H (V13-2339)	
B21-F515L (V13-2340)	
B21-F515M (V13-2341)	
B21-F515N (V13-2342)	
B21-F515P (V13-2343)	
B21-F515R (V13-2344)	
B21-F515S (V13-2345)	
B21-F515T (V13-2346)	
B21-F515U (V13-2347)	
b. RPV Instrumentation	
1) Level:	
B21-F507 (V13-2318)	
B21-F508 (V13-2319)	
B21-F509 (V13-2320)	
B21-F510 (V13-2321)	
B21-F511 (V13-2396)	
B21-F512 (V13-2323)	
2) Pressure:	
B21-F506 (V13-2317)	
B21-F508 (V13-2319)	
B21-F516A (V13-2348)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
11. Excess Flow Check Valves(r) (Continued)	NA
b. RPV Instrumentation (Continued)	
2) Pressure (Continued)	
B21-F516B (V13-2349)	
B21-F516C (V13-2388)	
B21-F517A (V13-2350)	
B21-F517B (V13-2389)	
B21-F517C (V13-2390)	
B21-F517D (V13-2391)	
B21-F518A (V13-2392)	
B21-F518B (V13-2393)	
G33-F583 (V13-2387)	
c. Core Spray Instrumentation	
E21-F500A (V13-2377)	
E21-F500B (V13-2378)	
d. HPCI Instrumentation	
E41-F500 (V13-2379)	
E41-F501 (V13-2380)	
E41-F502 (V13-2381)	
E41-F503 (V13-2382)	
e. RCIC Instrumentation	
E51-F503 (V13-2383)	
E51-F504 (V13-2384)	
E51-F505 (V13-2385)	
E51-F506 (V13-2386)	

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
11. Excess Flow Check Valves[r] (Continued)	NA
f. Recirculation Pump Instrumentation	
1) Flow	
Loop A: B31-F503A (V13-2359)	
B31-F504A (V13-2361)	
B31-F505A (V13-2363)	
B31-F506A (V13-2365)	
Loop B: B31-F503B (V13-2360)	
B31-F504B (V13-2362)	
B31-F505B (V13-2364)	
B31-F506B (V13-2366)	
2) Inlet Differential Pressure	
B31-F501A (V13-2351)	
B31-F501B (V13-2352)	
B31-F501C (V13-2353)	
B31-F501D (V13-2354)	
B31-F502A (V13-2355)	
B31-F502B (V13-2356)	
B31-F502C (V13-2357)	
B31-F502D (V13-2358)	
3) Pump Differential Pressure	
Pump A: B31-F510A (V13-2367)	
B31-F511A (V13-2369)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
11. Excess Flow Check Valves[r] Continued)	NA

f. Recirculation Pump Instrumentation (Continued)

3) Pump Differential Pressure (Continued)

Pump B: B31-F510B (V13-2368)
B31-F511B (V13-2370)

4) Seal Cavity Pressure

Pump A, #1 Seal: B31-F516A (V13-2375)
Pump A, #2 Seal: B31-F515A (V13-2373)

Pump B, #1 Seal: B31-F516B (V13-2376)
Pump B, #2 Seal: B31-F515B (V13-2374)

5) Pumps A and B Suction Pressure

B31-F512A (V13-2371)
B31-F512B (V13-2372)

g. Main Steam Flow Instrumentation:

Line A: B21-F501A (V13-2301)
B21-F502A (V13-2305)
B21-F503A (V13-2309)
B21-F504A (V13-2313)

Line B: B21-F501B (V13-2302)
B21-F502B (V13-2306)
B21-F503B (V13-2310)
B21-F504B (V13-2314)

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
11. Excess Flow Check Valves(r) (Continued)	NA
g. Main Steam Flow Instrumentation (Continued)	
Line C: B21-F501C (V13-2303)	
B21-F502C (V13-2307)	
B21-F503C (V13-2311)	
B21-F504C (V13-2315)	
Line D: B21-F501D (V13-2304)	
B21-F502D (V13-2308)	
B21-F503D (V13-2312)	
B21-F504D (V13-2316)	
12. HPCI Turbine Exhaust Drain Pot Drain To Suppression Chamber Reverse Stop Check Valve(n)	NA
E41-F022 (V11-2008)	
13. RCIC Turbine Exhaust Line Isolation Check Valve(n)	NA
E51-F001 (V11-2002)	
14. HPIC Turbine Exhaust Line Isolation Valve(n)	NA
E41-F021 (V11-2006)	
15. RCIC Barometric Condenser Vacuum Pump Discharge Stop Check Valve(n)	NA
E51-F002 (V8-2235)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
16. Combustible Gas Control System Return Line Relief Valves[r]	NA
Division I: T48-F016A (V22-2122)	
Division II: T48-F016B (V22-2121)	
17. Suppression Pool to Reactor Building Check Valves	NA
T23-F450A (V21-2013)	
T23-F450B (V21-2014)	
18. CRD Insert and Withdrawal Valves[o][r]	NA
C11-F120	
C11-F121	
C11-F122	
C11-F123	
19. Standby Liquid Control Reverse Flow Check Valves	NA
Inboard: C41-F007 (VR4-2012)	
Outboard: C41-F006 (VR4-2011)	
20. RWCU to Feedwater Reverse Flow Check Valve[n]	NA
G33-F120 (V8-2268)	
21. EECW Supply to Drywell Equipment Check Valves	NA
Division I: P44-F282A (V8-3888)	
Division II: P44-F282B (V8-3887)	

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	MAXIMUM ISOLATION TIME (Seconds)
D. Other Isolation Valves (Continued)	
22. TIP Purge System Isolation Valve	NA
C51-J009	
23. Control Rod Drive System Insert and Withdrawal Lines [(r)]	NA
115	
138	
24. Control Rod Drive Scram Discharge Volume	NA
C11-F010 (V8-2073)	
C11-F011 (V8-2886)	
C11-F120 (V8-3876)	
C11-F181 (V8-3877)	

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS

(a) The following is a summary of the parameters which will automatically actuate the Primary Containment Isolation Valve Groups. The instrumentation associated with these parameters is described in Specification 3.3.2.

1. Group 1 - Main Steam System

   Reactor Vessel Low Water Level - Level 1
   Main Steam Line Radiation - High
   Main Steam Line Flow - High
   Main Steam Line Tunnel Temperature - High
   Main Steam Line Pressure - Low
   Condenser Pressure - High
   Turbine Building Area Temperature - High

2. Group 2 - Reactor Water Sample System

   Reactor Vessel Low Water Level - Level 2
   Drywell Pressure - High
   Main Steam Line Radiation - High

3. Group 3 - Residual Heat Removal (RHR) System

   Reactor Vessel Low Water Level - Level 1
   Drywell Pressure - High

4. Group 4 - Residual Heat Removal Shutdown Cooling and Head Spray

   Reactor Vessel Low Water Level - Level 3
   Reactor Vessel Pressure - High, Shutdown Cooling Interlock

5. Group 5 - Core Spray System

   Reactor Vessel Low Water Level - Level 1
   Drywell Pressure - High

6. Group 6 - High Pressure Coolant Injection (HPCI) System

   HPCI Steam Line Flow - High
   HPCI Steam Supply Pressure - Low
   HPCI Turbine Exhaust Diaphragm Pressure - High
   HPCI Equipment Room Temperature - High

7. Group 7 - High Pressure Coolant Injection (HPCI) Vacuum Breakers

   Drywell Pressure - High with simultaneous HPCI
   Steam Supply Pressure - Low

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS (Continued)

8. Group 8 - Reactor Core Isolation Cooling (RCIC) System

RCIC Steam Line Flow - High
RCIC Steam Supply Pressure - Low
RCIC Turbine Exhaust Diaphragm Pressure - High
RCIC Equipment Room Temperature - High

9. Group 9 - Reactor Core Isolation Cooling (RCIC) Vacuum Breakers

Drywell Pressure - High with simultaneous RCIC
Steam Supply Pressure - Low

10. Group 10 - Reactor Water Cleanup (RWCU) System (Inboard)

RWCU Differential Flow - High
RWCU Area Temperature - High
RWCU Area Ventilation Differential Temperature - High
Reactor Vessel Low Water Level - Level 2

11. Group 11 - Reactor Water Cleanup (RWCU) System (Outboard)

SLCS Initiation (not a containment isolation signal)
RWCU Differential Flow - High
RWCU Area Temperature - High
RWCU Area Ventilation Differential Temperature - High
Reactor Vessel Low Water Level - Level 2

12. Group 12 - Torus Water Management System (TWMS)

Reactor Vessel Low Water Level - Level 2
Drywell Pressure - High

13. Group 13 - Drywell Sumps

Reactor Vessel Low Water Level - Level 3
Drywell Pressure - High

14. Group 14 - Drywell and Suppression Pool Ventilation System

Reactor Vessel Low Water Level - Level 2
Drywell Pressure - High
Fuel Pool Ventilation Exhaust Radiation - High

15. Group 15 - Traversing In-Core (TIP) System

Reactor Vessel Low Water Level - Level 3
Drywell Pressure - High

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS (Continued)

15. Group 15 - Traversing In-Core (TIP) System (Continued)

    NOTE: Either of these signals initiate TIP withdrawal which results in automatic closure of the TIP Ball Valves when the TIP probe has entered the shield cask.

16. Group 16 - Nitrogen Inerting System

    Reactor Vessel Low Water Level - Level 2
    Drywell Pressure - High
    Fuel Pool Ventilation Exhaust Radiation - High

17. Group 17 - Recirculation Pump System

    Reactor Vessel Low Water Level - Level 2
    Drywell Pressure - High

18. Group 18 - Primary Containment Pneumatic Supply System

    Reactor Vessel Low Water Level - Level 2
    Drywell Pressure - High

(b) Penetrations associated with these valves are hydrostatically leak tested.

(c) RWCU Water Temperature - High automatically closes only G33-F004, outboard isolation, and not G33-F001, inboard isolation. This is not a containment isolation signal.

(d) Also closes automatically as a result of Torus Room Floor Drain Sump Level - High - High and Drywell Floor Drain Sump Level - High - High.

(e) These valves may be closed remotely from either the control room or from their respective local panels.

(f) Will automatically reposition as a result of the actuation of the LPCI Loop Selection Logic.

(g) Will automatically close when the corresponding RHR loop flow is greater than 2200 gpm.

(h) Will automatically close when the corresponding core spray loop flow is greater than approximately 635 gpm.

(i) Will automatically close when a) HPCI Turbine Steam Stop Valve E41-F067 closes or b) HPCI Turbine Steam Supply Isolation Valve E41-F001 closes.

(j) Will automatically close as a result of the condition listed in Note (i), above, as well as when HPCI flow is greater than 1200 gpm.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS (Continued)

(k) Will automatically close when a) RCIC Turbine Steam Stop Valve E51-F045 closes or b) RCIC Turbine Governor Trip and Throttle Valve E51-F059 closes.

(l) Will automatically close as a result of the conditions listed in Note (k) above, as well as when RCIC flow is greater than 120 gpm.

(m) These valves are actuated by remote manual key-locked switches and will cut the TIP cable and seal off the TIP guide tube when actuated. These valves are squib-fired.

(n) May be closed remotely as a secondary actuation mode to reverse flow.

(o) Valves realign automatically on a reactor scram signal.

(p) Thermal relief valves.

(q) Normally locked closed.

(r) Not subject to Type C leakage tests.

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3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 All suppression chamber - drywell vacuum breakers shall be closed and at least 10 vacuum breakers shall be OPERABLE.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one of the above required vacuum breakers inoperable for opening but known to be closed, restore the inoperable vacuum breaker 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 or more suppression chamber - drywell vacuum breakers open, close the open vacuum breaker(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 one of the closed position indicator of any suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the drywell to suppression chamber ΔP is maintained at greater than or equal to 0.5 psi for 1 hour without makeup within 24 hours and at least once per 15 days thereafter. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*The suppression chamber-drywell vacuum breakers may be manually opened for inerting the containment. All these vacuum breakers shall be in the closed position within 2 hours after inerting is completed.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.

3. At least once per 18 months by;

a) Verifying the opening setpoint, from the closed position, to be less than or equal to 0.5 psid, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

REACTOR BUILDING - SUPPRESSION CHAMBER VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 All Reactor Building - suppression chamber vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one Reactor Building - suppression chamber vacuum breaker inoperable for opening but known to be closed, restore the inoperable vacuum breaker 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 Reactor Building - suppression chamber vacuum breaker open, isolate the associated vacuum breaker line by closing the isolation valve 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 the position indicator of any Reactor Building-suppression chamber vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker to be closed at least once per 24 hours by visual inspection. Otherwise, declare the vacuum breaker inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Reactor Building - suppression chamber vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

   1. At least once per 31 days by:

      a) Cycling vacuum breaker through at least one complete test cycle of full travel.

      b) Verifying the position indicator OPERABLE by observing expected valve movement during the cycling test.

   2. At least once per 18 months by:

      a) Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of 0.5 psid.

      b) Visual inspection.

      c) Verifying the position indicator OPERABLE by performance of a CHANNEL CALIBRATION.

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3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT INTEGRITY 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. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All secondary containment equipment hatches and pressure relief doors are closed and sealed and one railroad bay access door is closed.

   2. At least one door in each access to the secondary containment is closed.

   3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in the closed position.

c. At least once per 18 months:

   1. Verifying that one standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 329 seconds, and

   2. Operating one standby gas treatment subsystem for 1 hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the secondary containment at a flow rate not exceeding 3000 cfm.

---

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITION FOR OPERATION

3.6.5.2 The secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 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 damper secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blank flange.

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 secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the damper to service after maintenance, repair, or replacement work is performed on the damper or its associated actuator, control or power circuit by cycling the damper through at least one complete cycle of full travel and verifying the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation damper actuates to its isolation position.

c. By verifying the isolation time to be within its limit when tested pursuant to Specification 4.0.5.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

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TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

DAMPER/VALVE FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building Ventilation Exhaust Damper T41-F008	5
2. Reactor Building Ventilation Exhaust Damper T41-F009	5
3. Reactor Building Ventilation Supply Damper T41-F010	5
4. Reactor Building Ventilation Supply Damper T41-F011	5

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

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

ACTION:

a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

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

   2. In Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both standby gas treatment subsystems inoperable in Operational Condition *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria 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 3000 scfm ± 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 3000 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

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. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 11.0 inches water gauge while operating the filter train at a flow rate of 3000 scfm ± 10%.

2. 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.

3. Verifying that the heaters dissipate at least 18 kW when tested in accordance with ANSI N510-1980.

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SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in in accordance with ANSI N510-1980 while operating the system at a flow rate of 3000 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria 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 3000 scfm ± 10%.

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3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: With one drywell and/or suppression chamber 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 drywell and suppression chamber 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 1150°F within 60 minutes. Maintain J 1150°F for at least 2 hours.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

2. 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.

3. 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.

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.2 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The drywell and suppression chamber oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

*See Special Test Exception 3.10.5.

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3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least the following independent residual heat removal service water (RHRSW) system subsystems, with each subsystem comprised of:

a. Two OPERABLE RHRSW pumps, and

b. An OPERABLE flow path capable of taking suction from the associated ultimate heat sink and transferring the water through one RHR heat exchanger,

shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3, two subsystems.

b. In OPERATIONAL CONDITIONS 4 and 5, the subsystem(s) associated with systems and components required OPERABLE by Specifications 3.4.9.1, 3.4.9.2, 3.9.11.1, and 3.9.11.2.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one RHRSW pump inoperable, restore the inoperable pump to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one RHRSW pump in each subsystem inoperable, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one RHRSW subsystem otherwise inoperable, restore the inoperable subsystem to OPERABLE status with at least one OPERABLE pump within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With both RHRSW subsystems otherwise inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

*Whenever both RHRSW 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

ACTION: (Continued)

c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required residual heat removal service water system subsystems shall be demonstrated OPERABLE 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.

EMERGENCY EQUIPMENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent emergency equipment cooling water (EECW) system subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE EECW pump, and

b. An OPERABLE flow path capable of removing heat from the associated safety-related equipment.

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

ACTION:

With an emergency equipment cooling water system subsystem 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 and declare the associated safety-related equipment inoperable and take the ACTION required by Specifications 3.4.9.1, 3.4.9.2, 3.5.1, 3.5.2, 3.6.2.2, 3.6.2.3, 3.9.11.1, and 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The emergency equipment cooling water system shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing nonsafety-related equipment actuates to its isolation position and the associated EECW pump automatically starts on an actuation test signal.

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PLANT SYSTEMS

EMERGENCY EQUIPMENT SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.3 Two independent emergency equipment service water (EESW) system subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE emergency equipment service water pump, and

b. An OPERABLE flow path capable of taking suction from the associated ultimate heat sink and transferring the water through the associated EECW heat exchanger.

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

ACTION:

With an emergency equipment service water system subsystem 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, and declare the associated safety-related equipment inoperable and take the ACTION required by Specifications 3.4.9.1, 3.4.9.2, 3.5.1, 3.5.2, 3.6.2.2, 3.6.2.3, 3.9.11.1, and 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The emergency equipment service water system shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying the EESW pump automatically starts upon receipt of an actuation test signal.

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DIESEL GENERATOR COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.4 The diesel generator cooling water subsystem associated with each diesel generator shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE diesel generator cooling water pump, and

b. An OPERABLE flow path capable of taking suction from the associated ultimate heat sink and transferring cooling water through the associated diesel generator heat exchanger.

APPLICABILITY: When the diesel generator is required to be OPERABLE.

ACTION:

With one or more diesel generator cooling water subsystems inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.4 Each of the above required diesel generator cooling water subsystems shall be demonstrated OPERABLE:

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

b. At least once per 18 months by verifying that each pump starts automatically upon receipt of a start signal for the associated diesel generator.

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.5 Two independent residual heat removal (RHR) reservoirs shall be OPERABLE with each reservoir comprised of:

a. A minimum water volume of 2,990,000 gallons, equivalent to a water level of 580 feet.

b. An average water temperature of less than or equal to 80°F.

c. At least one OPERABLE cooling tower.

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

ACTION:

With the requirements of the above specification not satisfied:

a. For one reservoir, declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.4.9.1, 3.4.9.2, 3.5.1, 3.5.2, 3.6.2.2, 3.6.2.3, 3.9.11.1 and 3.9.11.2, as applicable,

b. For both reservoirs:

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

   2. In OPERATIONAL CONDITIONS 4 or 5, declare the RHRSW system, the EECW system, the EESW system and the emergency diesel generators inoperable and take the ACTION required by Specifications 3.7.1.1, 3.7.1.2, 3.7.1.3 and 3.8.1.2.

   3. In Operational Condition *, declare the emergency diesel generators inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each RHR reservoir shall be determined OPERABLE at least once per:

a. 24 hours by verifying the average water temperature and water level to be within their limits.

b. 31 days by:

   1. Starting each cooling tower fan from the control room and operating the fan on slow speed and on fast speed,** each for at least 15 minutes.

   2. Verifying each reservoir cross-connect valve to be closed.

c. 92 days by cycling each reservoir cross-connect valve through at least one cycle of full travel.

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*When handling irradiated fuel in the secondary containment.
**Fast speed need not be tested during icing periods.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The control room emergency filtration system shall be OPERABLE with the system composed of:

a. The emergency makeup air filter train.

b. The emergency recirculation air filter train.

c. Two recirculation fans.

d. Two return and supply fans.

e. A flowpath capable of:

   1. Recirculating control room air.

   2. Supplying emergency makeup air to the control room.**

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with one control room emergency filtration system recirculation fan and/or one return and supply fan inoperable, restore the inoperable fan(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.

b. In OPERATIONAL CONDITION 4, 5 or *:

   1. With one control room emergency filtration system recirculation fan and/or one return and supply fan inoperable, restore the inoperable fan(s) to OPERABLE status within 7 days or initiate and maintain operation of the system with OPERABLE fans in the recirculation mode of operation.

   2. With the control room emergency filtration system inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment 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 The control room emergency filtration system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating fan operation from the control room, and establishing flow through the HEPA filters and charcoal adsorbers, and verifying that the system operates for at least 10 hours with the heaters OPERABLE.

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*When irradiated fuel is being handled in the secondary containment.

**Not applicable in the chlorine mode of operation.

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SURVEILLANCE REQUIREMENTS (Continued)

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

   1. Verifying that the system satisfies the in-place penetration testing acceptance criteria 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 flow rate is 1800 scfm ± 10% through the makeup filter and 3000 scfm ± 10% through the recirculation filter.

   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 1.0%; and

   3. Verifying a system flow rate of 3000 scfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1.0%.

e. At least once per 18 months by:

   1. Verifying that the pressure drop across the recirculation train and across the makeup train combined HEPA filters and charcoal adsorber banks are each less than 6 inches water gauge while operating the system at a flow rate of 3000 scfm ± 10%.

   2. Verifying that the system will automatically switch to the recirculation mode of operation on each of the below actuation test signals and verifying that on any one of the below recirculation mode actuation test signals, the system automatically switches to the recirculation mode of operation, the isolation valves close within 5 seconds and the control room is maintained at a positive pressure of at least 0.125 inch water gauge relative to the outside atmosphere during system operation at a flow rate less than or equal to 3000 scfm through the emergency recirculation filter:

      a) Control center inlet radiation monitor.
      b) Reactor Building ventilation exhaust RAD. monitor
      c) Radwaste Building ventilation exhaust RAD. monitor.
      d) Turbine Building ventilation exhaust RAD. monitor.
      e) Fuel pool ventilation exhaust radiation monitor.
      f) Low reactor water level.
      g) High drywell pressure.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that on the chlorine mode actuation signal, the system automatically switches to the chlorine mode of operation, the isolation valves close within 4 seconds, and a minimum of 1200 scfm emergency recirculation is established.

4. Verifying that the makeup filter train heaters dissipate 12.0 ± 2.0 kW when tested in accordance with ANSI N510-1975.

f. After each complete or partial replacement of a train HEPA filter bank by verifying that the train HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 while operating the system at a flow rate of 1800 scfm ± 10% for the makeup train and 3000 scfm for the recirculation train.

g. After each complete or partial replacement of a train charcoal adsorber bank by verifying that the train charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 1800 scfm ± 10% for the makeup train and 3000 scfm ± 10% for the recirculation train.

3/4.7.3 SHORE BARRIER PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.3 The shore barrier shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With the shore barrier inoperable prepare and submit to the Commission within 7 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

a. Explanation of the damage to the shore barrier,

b. The actions taken and/or proposed actions to restore the shore barrier to OPERABLE status,

c. Evaluation of and justification for continued plant operation.

SURVEILLANCE REQUIREMENTS

4.7.3 The shore barrier shall be determined to be OPERABLE by visual inspection and survey:

a. At least once per 12 months.

b. Within 7 days after a severe storm in which the crest elevation of incident waves at the shore line exceeds the top of the shore barrier (583'0").

3/4 7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of 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 HPCI 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

4.7.4 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 reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

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*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.

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 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.

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*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.

3/4.7.5 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.5 All* hydraulic and mechanical snubbers shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3. 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 4.7.5g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.5 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

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*As described in the bases.

No. of Inoperable Snubbers of Each Type on Any System per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.5f. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 72 hours for accessible areas and 6 months for inaccessible areas following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.5f., 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 4.7.5-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.5f. 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 4.7.5-1. 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 points falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be

plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted fall above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

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

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

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

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

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

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

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.5e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

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

i. Snubber Seal Replacement Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

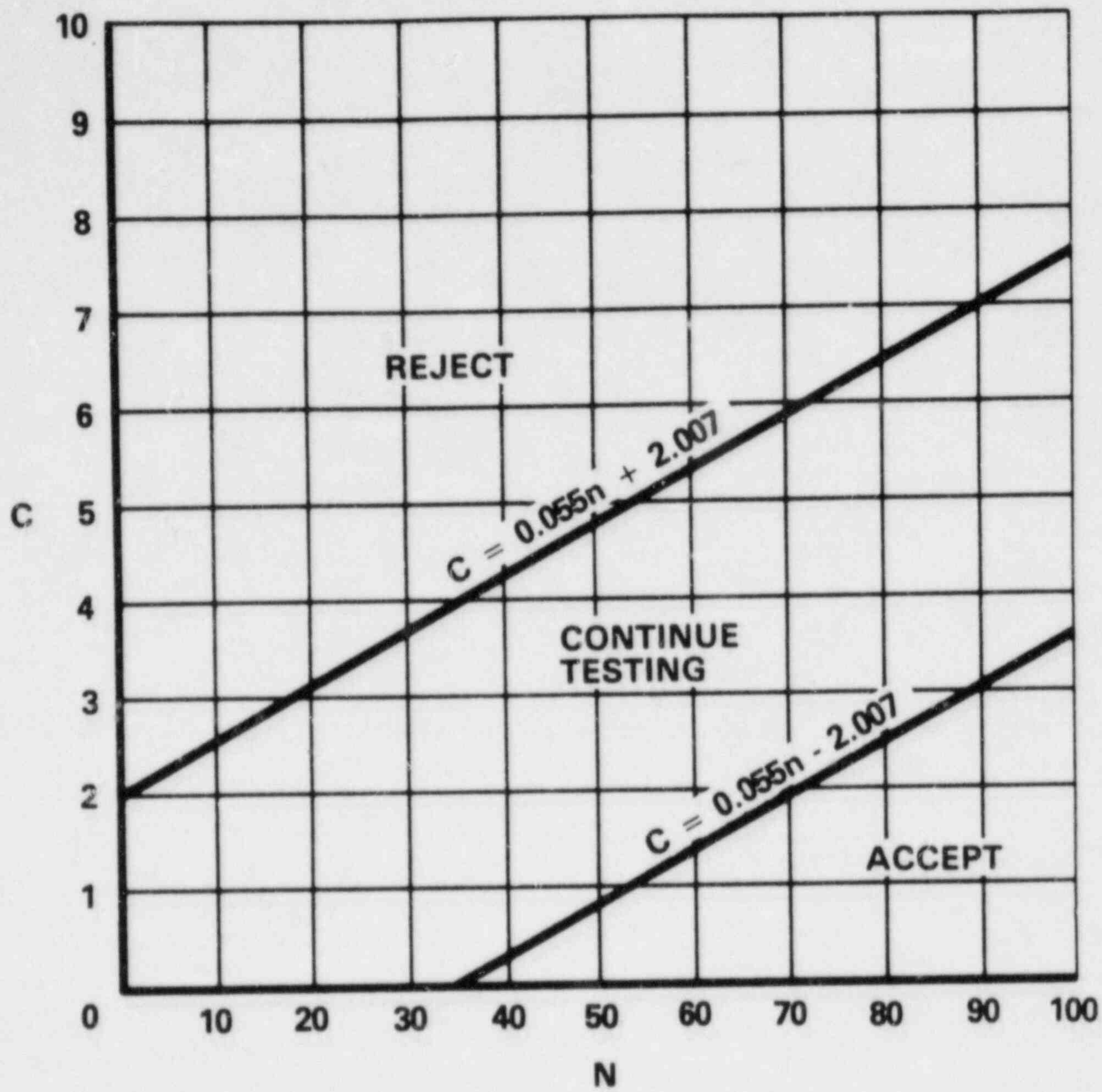

FIGURE 4.7.5-1
SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

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

a. The licensee, or

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

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

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

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive material:

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

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

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

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

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3/4.7.7 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The fire suppression water system shall be OPERABLE with:

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

b. The general service water intake structure water level $\geq$ 556 feet, and

c. An OPERABLE flow path capable of taking suction from the general service water intake structure 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 spray system required to be OPERABLE per Specifications 3.7.7.2, 3.7.7.5, and 3.7.7.6.

APPLICABILITY: At all times.

ACTION:

a. With one pump inoperable, restore the inoperable pump to OPERABLE status within 7 days or provide an alternate backup pump. 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

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the minimum water supply level.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each fire suppression pump develops at least 2500 gpm at a discharge pressure head of 145 psig,

   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 105 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 8, Section 16 of the Fire Protection Handbook, 15th Edition, published by the National Fire Protection Association.

4.7.7.1.2 The diesel-driven fire suppression pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying the fuel storage tank contains at least 150 gallons of fuel.

   2. Starting the diesel driven pump from ambient conditions and operating 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.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.7.1.3 The 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 battery is above the plates,

   2. The battery specific gravity, corrected to 77°F, is greater than or equal to 1.200,

   3. The battery voltage is greater than or equal to 24 volts, and

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

   1. The battery 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 anticorrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.2 The following spray and sprinkler systems shall be OPERABLE:

AREA		ELEVATION	TYPE
a.	Reactor Building		
	1. Torus Room	560'	Wet Pipe Sprinkler
	2. Basement NE Corner Room	540'	Wet Pipe Sprinkler
	3. HPCI Turbine and Pump Room	540'	Wet Pipe Sprinkler
	4. First Floor, Railroad Bay	583'	Wet Pipe Sprinkler
	5. Second Floor, Cable Trays	613'	Wet Pipe Sprinkler
	6. Fourth Floor, MG Sets	641'6"	Wet Pipe Sprinkler
b.	Auxiliary Building		
	1. Basement	551' and 562'	Wet Pipe Sprinkler
	2. Mezzanine and Cable Tray Area	583' - 603'	Wet Pipe Sprinkler
	3. Ventilation Equipment Area	677'	Manual Flooding System
	4. Cable Spreading Room	630'6"	Manual Sprinkler
	5. Corridor	562'	Wet Pipe Sprinkler
c.	RHR Complex		
	1. Fuel Oil Storage Tank Rooms	-	Wet Pipe Sprinkler
d.	General Service Water Pumphouse		
	1. Diesel Fire Pump Room	-	Wet Pipe Sprinkler

APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; 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

4.7.7.2 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 by a visual inspection of the sprinkler header to verify its integrity.

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.3 The following low pressure CO_2 systems shall be OPERABLE:

a. Emergency diesel generators, RHR complex.

b. Standby gas treatment system charcoal filters, Auxiliary Building, elevation 677'6".

c. Cable tray area, Auxiliary Building, elevation 631'.

d. Outside Division II switchgear room, Auxiliary Building, elevation 643'6".

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within 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

4.7.7.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4.7.7.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 50% full for systems a and b above and greater than 40% full for systems c and d above, and pressure to be greater than 250 psig but less than 315 psig for all of the systems.

b. At least once per 18 months by verifying:

1. The system, including associated ventilation system fire dampers and fire door release mechanisms, actuates, manually and/or automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

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HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.4 The following Halon systems shall be OPERABLE with the storage tanks of either the main bank or the reserve bank having at least 95% of the main bank or the reserve bank full charge weight and 90% of the main bank or the reserve bank full charge pressure:

a. Relay room, elevation 613'6".

b. Cable spreading room, elevation 630'6".

c. Computer room, and under floor, elevation 655'6".

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

4.7.7.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

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

c. At least once per 18 months by:

   1. Verifying the system, including associated ventilation system fire dampers and fire door release mechanisms, actuates, manually and automatically, upon receipt of a simulated actuation signal, and

   2. Performance of a puff test through all headers and nozzles to assure no blockage.

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.7.5 The fire hose stations shown in Table 3.7.7.5-1 shall be OPERABLE.

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

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7.7.5-1 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye 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 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

4.7.7.5 Each of the fire hose stations shown in Table 3.7.7.5-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose 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.

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 3.7.7.5-1

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
a. Reactor Building		
1. Fire hose at top of stairway in northwest aux building	736'	RB-1
2. Fire hose at northwest corner by elevator	684'-6"	RB-2
3. Fire hose at southwest corner	684'-6"	RB-3
4. Fire hose at northeast stairway	684'-6"	RB-4
5. Fire hose in southeast walkway	684'-6"	RB-5
6. Fire hose at northwest corner outside elevator	659'-6"	RB-6
7. Fire hose at northeast corner in stairway	659'-6"	RB-7
8. Fire hose at southwest corner at stairway	659'-6"	RB-8
9. Fire hose at southeast corner at stairway	659'-6"	RB-9
10. Fire hose at northeast corner at stairway	641'-6"	RB-10
11. Fire hose at northwest corner at stairway by elevator	641'-6"	RB-11
12. Fire hose at southwest corner at stairway	641'-6"	RB-12
13. Fire hose at southeast corner at stairway	641'-6"	RB-13
14. Fire hose at northwest corner near elevator	613'-6"	RB-14
15. Fire hose at southwest corner at bottom of stairway	613'-6"	RB-15
16. Fire hose near drywell instrument monitoring rack (east walkway)	613'-6"	RB-16

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TABLE 3.7.7.5-1 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
17. Fire hose in the northeast corner	613'-6"	RB-17
18. Fire hose at southeast corner by aux bldg access	613'-6"	RB-18
19. Fire hose at northwest corner near elevator	583'-6"	RB-19
20. Fire hose at northeast corner near stairway	583'-6"	RB-20
21. Fire hose at railroad bay	583'-6"	RB-21
22. Fire hose at southeast corner near stairway	583'-6"	RB-22
23. Fire hose at entrance to containment (southwest)	583'-6"	RB-23
24. Fire hose at northwest corner near elevator	562'-0"	RB-24
25. Fire hose at northeast corner near stairway	562'-0"	RB-25
26. Fire hose at southwest corner near stairway	562'-0"	RB-26
27. Fire hose at southeast corner near stairway	562'-0"	RB-27
28. Fire hose at northwest corner near stairway	540'-0"	RB-28
29. Fire hose at northeast corner near stairway	540'-0"	RB-29
30. Fire hose at southwest corner near stairway	540'-0"	RB-30
31. Fire hose at southeast corner near stairway	540'-0"	RB-31
32. Fire hose in HPCI room	540'-0"	RB-32
33. Fire hose in CRD pump room	562'-0"	RB-33

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TABLE 3.7.7.5-1 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
b. Auxiliary Building		
1. Fire hose at southwest corner in control center air conditioning equipment room	677'-6"	AB-1
2. Fire hose at northwest corner in ventilation equipment area	677'-6"	AB-2
3. Fire hose at southwest wall in ventilation equipment area	677'6"	AB-3
4. Fire hose at north side in ventilation equipment area	659'-6"	AB-4
5. Fire hose at south side in ventilation equipment area	659'-6"	AB-5
6. Fire hose outside control room near center stairway	643'-6"	AB-6
7. Fire hose outside cable spreading room in stairway from control room	630'-6"	AB-7
8. Fire hose south wall cable tray room near stairway	630'-6"	AB-8
9. Fire hose near column line H-12	613'-6"	AB-9
10. Fire hose in walkway from reactor building	613'-6"	AB-10
11. Fire hose in stairway from relay room to lower cable tray area	613'-6"	AB-11
12. Fire hose at southeast corner by RBCCW heat exchanger	583'-6"	AB-12
13. Fire hose at column G, 14 RBCCW pump area	583'-6"	AB-13
14. Fire hose near compressor receiver for Division II	551'-0"	AB-14
15. Fire hose near compressor receiver for Division I	551'-0"	AB-15

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TABLE 3.7.7.5-1 (Continued)

FIRE HOSE STATIONS

LOCATION		ELEVATION	HOSE RACK#
c.	Residual Heat Removal (RHR) Complex		
1.	Fire hose at top of stairway to RHR-1 switchgear room	617'-0"	RR-1
2.	Fire hose at top of stairway to RHR-2 switchgear room	617'-0"	RR-2
3.	Fire hose in RHR-1 near diesel generator service water pump	590'-0"	RR-3
4.	Fire hose in RHR-2 near diesel generator service water pump	590'-0"	RR-4
5.	Fire hose in RHR-1 near diesel generator #12	590'-0"	RR-5
6.	Fire hose in RHR-2 near diesel generator #13	590'-0"	RR-6
7.	Fire hose in RHR-1 near diesel generator #11	590'-0"	RR-7
8.	Fire hose in RHR-2 near diesel generator #14	590'-0"	RR-8

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.7.6 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.6-1 shall be OPERABLE.

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

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7.7.6-1 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

4.7.7.6 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.6-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months, during March, April or May and during September, October or November, by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure 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 3.7.7.6-1

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION	HYDRANT NUMBER
a. Between the RHR complex and the Reactor Building	9
b. Southwest of the Reactor Building	10
c. Southwest of the Reactor Building	11
d. Southeast of the Reactor Building	12

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3/4.7.8 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.8 All fire rated assemblies, including walls, floor/ceilings, cable tray enclosures and other fire barriers, separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area, and all sealing devices in fire rated assembly penetrations including fire doors, fire dampers, cable, piping and ventilation duct penetration seals and ventilation seals, shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour establish a continuous fire watch on at least one side of the affected assembly(s) and/or sealing device(s) or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly(s) and sealing device(s) and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.8.1 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% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. The position of each locked-closed fire door at least once per 7 days.

c. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

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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 take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.7.9 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. At least once per 92 days and during each COLD SHUTDOWN, by cycling each turbine bypass valve through at least one complete cycle of full travel, and

b. 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 TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to 300 milliseconds.

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent onsite A.C. electrical power sources, Division I and Division II, each consisting of two emergency diesel generators, each diesel generator with:

1. A separate day fuel tank containing a minimum of 210 gallons of fuel,

2. A separate fuel storage system containing a minimum of 35,280 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or both offsite circuits of the above required A.C. electrical power sources inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours; demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1. and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter.

b. With one or both diesel generators in one of the above required onsite A.C. electrical power divisions inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1. and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore the inoperable division 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.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With one or both diesel generators in one of the above required onsite A.C. electrical power divisions inoperable, in addition to ACTION b, above, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining onsite A.C. electrical power division as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With both of the above required onsite A.C. electrical power divisions inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter; restore at least one of the above required inoperable divisions 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 of the above required divisions 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.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability.

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 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual.
   b) Simulated loss-of-offsite power by itself.
   c) Simulated loss-of-offsite power in conjunction with an ESF actuation test signal.
   d) An ESF actuation test signal by itself.

5. Verifying the diesel generator is synchronized, loaded to greater than or equal to 2850 kW in less than or equal to 150 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 225 psig.

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

c. At least once per 92 days by removing accumulated water from the fuel storage tank (s).

d. At least once per 92 days and from new fuel oil prior to addition to the storage tanks by obtaining a sample in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

1. As soon as sample is taken or from new fuel prior to addition to the storage tank, as applicable, verify in accordance with the tests specified the ASTM-D975-77 that the sample has:

a) A water and sediment content of less than or equal to 0.05 volume percent.

b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

c) A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to 0.7889 but less than or equal to 0.8251 or an API gravity @ 60°F of greater than or equal to 30 degrees but less than or equal to 40 degrees.

2. Within one week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 mL when tested in accordance with ASTM-D2274-70.

3. Within two weeks after obtaining the sample, verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137, Position 2.a, are met when tested in accordance with ASTM-D975-77.

e. 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 1666 kW while maintaining engine speed $\geq$ 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 2850 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

SURVEILLANCE REQUIREMENTS (Continued)

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

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the autoconnected loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 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) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the 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.

7. Verifying that all automatic diesel generator trips, except overspeed, generator differential, low lube oil pressure, crankcase overpressure, and failure to start are automatically bypassed upon loss of voltage on emergency start signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 3135 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2850 kW. 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.e.4.b).*

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW.

10. Verifying the diesel generator's capability to:

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

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

    c) Be restored to its standby status.

11. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

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

    a) 4160-volt ESF bus lockout.

    b) Differential trip.

    c) Shutdown relay trip.

*If Surveillance Requirement 4.8.1.1.2.e.4.b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 2850 kW for 1 hour or until operating temperature has stabilized.

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

g. 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 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 in a Special Report 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

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 100 Valid Tests*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31 day test frequency.

A.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One onsite A.C. electrical power source, Division I or Division II, consisting of two emergency diesel generators, each diesel generator with:

   1. A day fuel tank containing a minimum of 210 gallons of fuel.

   2. A fuel storage system containing a minimum of 35,280 gallons of fuel.

   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 secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 20 feet 6 inches above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

*When handling irradiated fuel in the secondary containment.

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3/4.8.2 D.C. SOURCES

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. 130 VDC Battery 2A-1.
   2. 130 VDC Battery 2A-2.
   3. Two 130 VDC full capacity chargers.

b. Division II, consisting of:
   1. 130 VDC Battery 2B-1.
   2. 130 VDC Battery 2B-2.
   3. Two 130 VDC full capacity chargers.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With either Division I or Division II 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.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 130-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 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above 150 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} ohm, and
   3. The average electrolyte temperature of ten of the connected cells is above 60°F.

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

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

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

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

4. The battery charger will supply at least 100 amperes at a minimum of 129 volts for at least 4 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 while maintaining the battery terminal voltage greater than or equal to 105 or 210 volts, as applicable:

a) Batteries 2PA and 2PB greater than or equal to 180 amperes during the initial 48 seconds of the test.

b) Batteries 2PA and 2PB greater than or equal to 52 amperes during the remainder of the 4 hour test.

e. at least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when sub-jected to a performance discharge test. At this 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

Parameter	CATEGORY A(1)	CATEGORY B(2)	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(4)	> 2.07 volts
Specific Gravity (5)	$\geq$ 1.195(6)	$\geq$ 1.190 Average of all connected cells > 1.200	Not more than 0.020 below the average of all connected cells Average of all connected cells $\geq$ 1.190(6)

(1)For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2)For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3)Any Category B parameter not within its allowable value indicates an inoperable battery.

(4)May be corrected for average electrolyte temperature.

(5)Corrected for electrolyte temperature and level.

(6)Or battery charging current is less than 2 amperes when on float charge.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division I or Division II of the D.C. electrical power sources system shall be OPERABLE with:

a. Division I, consisting of:

1. 130 VDC Battery 2A-1.
2. 130 VDC Battery 2A-2.
3. Two 130 VDC full capacity chargers.

b. Division II, consisting of:

1. 130 VDC Battery 2B-1.
2. 130 VDC Battery 2B-2.
3. Two 130 VDC full capacity chargers.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With both of the above required Division I and Division II battery and/or charger D.C. electrical power sources inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, and operations with a potential for draining the reactor vessel.

b. 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 secondary containment.

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3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized with tie breakers open between redundant buses within the unit:

a. A.C. power distribution:

1. Division I, consisting of:

a) 4160V RHR Complex Busses 11EA and 12EB.
b) 4160V Reactor Building Busses 64B and 64C.
c) 480V RHR Complex Busses 72EA and 72EB.
d) 480V Reactor Building Busses 72B and 72C.
e) 120V Division I I&C Power Supply Unit, MPU 1.

2. Division II, consisting of:

a) 4160V RHR Complex Busses 13EC and 14ED.
b) 4160V Reactor Building Busses 65E and 65F.
c) 480V RHR Complex Busses 72EC and 72ED.
d) 480V Reactor Building Busses 72E and 72F.
e) 120V Division II I&C Power Supply Unit, MPU 2.

b. D.C. power distribution:

1. Division I, consisting of:

a) 130-volt D.C. Distribution Cabinet 2PA-2.
b) 260-volt D.C. MCC 2PA-1.

2. Division II, consisting of:

a) 130-volt D.C. Distribution Cabinet 2PB-2.
b) 260-volt D.C. MCC 2PB-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one of the above required A.C. distribution system divisions not energized, reenergize 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.

b. With one of the above required D.C. distribution system divisions not energized, reenergize 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.

SURVEILLANCE REQUIREMENTS

4.8.3.1.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/cabinets.

4.8.3.1.2 The A.C. power distribution system swing bus automatic throwover scheme shall be demonstrated OPERABLE at least once per 31 days by manually opening position 3C bus 72C and verifying that the automatic transfer scheme operates.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, Division I or Division II of the power distribution system shall be energized with:

a. A.C. power distribution:

1. Division I, consisting of:

   a) 4160V RHR Complex Busses 11EA and 12EB.
   b) 4160V Reactor Building Busses 64B and 64C.
   c) 480V RHR Complex Busses 72EA and 72EB.
   d) 480V Reactor Building Busses 72B and 72C.
   e) 120V Division I I&C Power Supply Unit, MPU 1.

2. Division II, consisting of:

   a) 4160V RHR Complex Busses 13EC and 14ED.
   b) 4160V Reactor Building Busses 65E and 65F.
   c) 480V RHR Complex Busses 72EC and 72ED.
   d) 480V Reactor Building Busses 72E and 72F.
   e) 120V Division II I&C Power Supply Unit, MPU 2

b. D.C. power distribution:

1. Division I, consisting of:

   a) 130-volt D.C. Distribution Cabinet 2PA-2.
   b) 260-volt D.C. MCC 2PA-1,

2. Division II, consisting of:

   a) 130-volt D.C. Distribution Cabinet 2PB-2.
   b) 260-volt D.C. MCC 2PB-1.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than Division I or Division II of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With less than Division I or Division II of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2.1 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/cabinets.

4.8.3.2.2 The A.C. power distribution system swing bus automatic throwover scheme shall be demonstrated OPERABLE at least once per 31 days by manually opening position 3C bus 72C and verifying that the automatic transfer scheme operates.

*When handling irradiated fuel in the secondary containment.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be deenergized*:

a. Circuit Number 6 in panel 72B-2D
b. Circuit Numbers 1, 2, 3, 4, 5, 15, 16, 17, 18 in panel R1R
c. Circuit Number 5 in panel H11-P907B
d. Circuit Number 4 in panel H21-P552
e. Circuit Number 1 in panel H11-P901
f. Circuit Number 1 in panel H11-P906C

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With any of the above required circuits energized, open the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be deenergized at least once per 24 hours** by verifying that the associated circuit breakers are in the off position.

*Except during entry into the drywell.

**Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 All primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 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 3.8.4.2-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system, and

   1. For 4.16-kV circuits, deenergize the 4.16-kV circuit(s) by tripping the associated circuit breaker(s) within 72 hours and verify the circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 480-volt circuit devices, remove the inoperable device(s) from service by racking out or removing the device within 72 hours and verify the inoperable device(s) to be racked out or removed at least once per 7 days thereafter.

   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 circuit breakers tripped or to 480-volt circuits which have the inoperable circuit breaker racked out.

SURVEILLANCE REQUIREMENTS

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.2-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the 4.16-kV circuits are OPERABLE by 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.

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.2-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP SETPOINT (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
1. 4.16-kV Circuit Breaker					
B-31-P003A Recirc Pump A Generator Field Breaker	GE	AC-50 (K9A, K22A)	1440A	NA	B31-C001A Recirc Pump A Motor
2. 4.16-kV Circuit Breaker					
B31-P003B Recirc Pump B Generator Field Breaker	GE	AC-50 (K9B, K22B)	1440A	NA	B31-C001B Recirc Pump B Motor
3. 480-V A.C.			TRIP OR FUSE RATING (A)		
30 A fuse disconnect (MCC 72E-3A)	Bussmann (FRS)	72E-3A-1A(R)	15 A	N.A.	G1101-C001B Drywell floor drain sump 72 pump
72E-3A-1A(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-3A-1A(R)	15 A	N.A.	
30A fuse disconnect (MCC 72E-3A)	Bussmann (FRS)	72E-3A-1B(R)	15 A	N.A.	G1101-C006B Drywell equipment drain sump 71 pump
72E-3A-1B(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-3A-1B(R)	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72E-3A)	Bussmann (FRS)	72E-3A-2B(R)	15 A	N.A.	G3352-F102 (V8-2251) common recirculation line valve
72E-3A-2B(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-3A-2B(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72E-3A)	Bussmann (FRS)	72E-3A-2C(R)	15 A	N.A.	G3352-F106 (V8-2249) recirculation line B valve
72E-3A-2C(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-3A-2C(R)	15 A	N.A.	
15 A Circuit breaker (MCC 72E-3A)	ITE (HE3B015)	72E-3A-1C(R)	15 A	N.A.	B3101-C001B recirculation pump B motor heater
72E-3A-1C(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-3A-1C(R)	15 A	N.A.	
30A fuse disconnect (MCC 72E-3A)	Bussman (FRS)	72E-3A-2A(R)	15 A	N.A.	G3351-F100 (V8-2250) recirculation line A valve
72E-3A-2A(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-3A-2A(R)	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2D(R)	20 A	N.A.	B3101-F023B (V8-2002) recirculation pump B suction valve
72B-4A-2D(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2D(R)	20 A	N.A.	
60 A fuse disconnect (MCC 72E-5B)	Bussmann (FRS)	72E-5B-1B	60 A	N.A.	T4700-C011 drywell cooling fan 11
72E-5B-1B (fuse box R1600S004E)	Bussmann (FRS)	72E-5B-1B	60 A	N.A.	
60 A fuse disconnect (MCC 72E-5B)	Bussmann (FRS)	72E-5B-1A(R)	60 A	N.A.	T4700-C012 drywell cooling fan 12
72E-5B-1A(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-5B-1A(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72E-5B)	Bussmann (FRS)	72E-5B-1A	60 A	N.A.	T4700-C010 drywell cooling fan 10
72E-5B-1A (fuse box R1600S004E)	Bussmann (FRS)	72E-5B-1A	60 A	N.A.	
60 A fuse disconnect (MCC 72E-5B)	Bussmann (FRS)	72E-5B-2A(R)	60 A	N.A.	T4700-C014 drywell cooling fan 14
72E-5B-2A(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-5B-2A(R)	60 A	N.A.	

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
15 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-4D(F)	15 A	N.A.	E5150-F007 (V17-2030) RCIC steam line inboard isolation valve
72F-4A-4D(F) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-4D(F)	15 A	N.A.	
30 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-5B(F)	15 A	N.A.	P4400-F608 (V8-2487) RBCCW to drywell equipment sump HX inlet valve
72F-4A-5B(F) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-5B(F)	15 A	N.A.	
30 A fuse disconnect (MCC 72F-4A)	Bussmann	72F-4A-3B(R)	15 A	N.A.	G1154-F600 (V9-2044) drywell floor drain valve motor
72F-4A-3B(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-3B(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-2B(R)	15 A	N.A.	G1154-F018 (V9-2022) drywell equipment drain sump discharge valve motor
15 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-2A	15 A	N.A.	P4400-F615 EECW Return from Drywell Isolation Valve
72F-4A-2A (fuse box R16005005G)	Bussman (FRS)	72F-4A-2A	15 A	N.A.	

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
72F-4A-2A (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-2A	15 A	N.A.	
72F-4A-2B(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-2B(R)	15 A	N.A.	
100 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-2A(R)	80 A	N.A.	T4700-C004 drywell cooling fan 4
72F-4A-2A(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-2A(R)	80 A	N.A.	
30 A fuse disconnect (MCC 72C-F)	Bussmann (FRS)	72C-F-5A	20 A	N.A.	B3101-F031B (V8-2004) recirculation pump B discharge valve
72C-F-5A (fuse box R1600S003J)	Bussmann (FRS)	72C-F-5A	20 A	N.A.	
30 A fuse disconnect (MCC 72C-F)	Bussmann (FRS)	72C-F-1B	20 A	N.A.	B3101-F031A (V8-2003) recirculation pump A discharge valve
72C-F-1B (fuse box R1600S003J)	Bussmann (FRS)	72C-F-1B	20 A	N.A.	
100 A fuse disconnect (MCC 72E-5A)	Bussmann (FRS)	72E-5A-1A	80 A	N.A.	T4700-C003 drywell cooling fan 3
72E-5A-1A (fuse box R1600S004E)	Bussmann (FRS)	72E-5A-1A	80 A	N.A.	

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
60 A fuse disconnect (MCC 72E-5B)	Bussmann (FRS)	72E-5B-1B(R)	60 A	N.A.	T4700-C013 drywell cooling fan 13
72E-5B-1B(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-5B-1B(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-4D(R)	40 A	N.A.	E1150-F608 (V8-3407) reactor recirculation extractor isolation to RHR valve
72F-4A-4D(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-4D(R)	40 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2A	15 A	N.A.	G1101-C001A drywell floor drain sump 72 pump
72B-4A-2A (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2A	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2B	15 A	N.A.	G1101-C006A drywell equipment drain sump 71 pump
72B-4A-2B (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2B	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72B-2A)	Bussmann (FRS)	72B-2A-4B	15 A	N.A.	P4400-F616 (V8-3890) EECW return drywell
72B-2A-4B (fuse box R1600S002D)	Bussmann (FRS)	72B-2A-4B	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-1D(R)	15 A	N.A.	G3352-F101 (V8-2254) vessel drain line recirculation valve
72B-4A-1D(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1D(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-1A(R)	15 A	N.A.	G1154-F015 (V9-2021) drywell recirculation equipment drains sump valve
72B-4A-1A(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1A(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2C(R)	15 A	N.A.	P5000-F604 (V5-2007) drywell station air inboard isolation valve
72B-2D-Pos 6	Bussman (FRS)	72B-2D-6	15 A	N.A.	F1700-E006 Under Vessel Equipment Handling Platform

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
72B-4A-2C(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2C(R)	15 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-2B	60 A	N.A.	T4700-C007 drywell cooling fan 7
72B-4C-2B (fuse box R1600S002C)	Bussmann (FRS)	72B-4C-2B	60 A	N.A.	
60 A fuse disconnect (MCC 72B04C)	Bussmann (FRS)	72B-4C-1A(R)	60 A	N.A.	T4700-C008 drywell cooling fan 8
72B-4C-1A(R) (fuse box R1600S002C)	Bussmann (FRS)	72B-4C-1A(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-1D	60 A	N.A.	T4700-C005 drywell cooling fan 5
72B-4C-1D (fuse box R1600S002C)	Bussmann (FRS)	72B-4C-1D	60 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-2A	60 A	N.A.	T4700-C006 drywell cooling fan 6
72B-4C-2A (fuse box R1600S002C)	Bussmann (FRS)	72B-4C-2A	60 A	N.A.	

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-4A	20 A	N.A.	E4150-F002 (V17-2020) HPCI steam supply inboard isolation valve 1
72C-3A-4A (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-4A	20 A	N.A.	
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-3D	15 A	N.A.	E1150-F022 (V8-2172) RHR head spray inboard isolation valve
72C-3A-3D (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-3D	15 A	N.A.	
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-4B	15 A	N.A.	B2103-F016 (V17-2009) main steam line drains inboard isolation valve
72C-3A-4B (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-4B	15 A	N.A.	
30 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5B	15 A	N.A.	G3352-F001 (V8-2252) cleanup supply inboard isolation valve

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
[illegible] A.C. (Contin[illegible]d)					
72B-3A-5B (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-5B	15 A	N.A.	
100 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-1A(R)	80 A	N.A.	T4700-C001 drywell cooling fan 1
72B-3A-1A(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-1A(R)	80 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-1B(R)	60 A	N.A.	T4700-C009 drywell cooling fan 9
72B-4C-1B(R) (fuse box R1600S002C)	Bussmann (FRS)	72B-4C-1B(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-3C	40 A	N.A.	E1150-F009 (V8-2091) RHR suction cooling inboard isolation

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
72B-4A-1C(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1C(R)	20 A	N.A.	
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-1A	15 A	N.A.	P4400-F614 (V8-3058) drywell penetration cooling jacket inlet valve
72C-3A-1A (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-1A	15 A	N.A.	
30 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5B(R)	15 A	N.A.	T4803-F601 (VR3-3011) nitrogen supply drywell inboard isolation valve
72B-3A-5B(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-5B(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5C(R)	15 A	N.A.	T4803-F602 (VR3-3024) drywell inboard isolation valve
72B-3A-5C(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-5C(R)	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
100 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-9C	80 A	N.A.	T4700-C002 drywell cooling fan 2
72C-3A-9C (fuse box R1600S003H)	Bussmann	72C-3A-9C	[illegible]	N.A.	
30 A circuit breaker (MCC 72B-4A)	ITE (HE3B015)	72B-4A-1A	15 A	N.A.	B3101-C001A recirculation Pump A motor heater
72B-4A-1A (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1A	6.25 A	N.A.	
15 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5A(R)	15 A	N.A.	T4901-F601 (V4-2080) nitrogen supply inboard isolation valve
72B-3A-5A(R) (fuse box R1600S002B)	Bussmann (FRS)	72B-3A-5A(R)	15 A	N.A.	
72E-5A-5E (fuse box R1600S004E)	Bussmann (FRS)	72E-5A-5E	15 A	N.A.	
15 A fuse disconnect (MCC 72E-5A)	Bussmann (FRS)	72E-5A-5E	15 A	N.A.	T4901-F602 (V4-2188) inboard isolation valve

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TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
30 A fuse disconnect (MCC 72E-5A)	Bussman (FRS)	72E-5A-5E	15 A	N.A.	T4901-F602 (VA-2188) inboard isolation valve
72E-5A-5E (fuse box R1600S004E)	Bussmann (FRS)	72E-5A-5E	15 A	N.A.	T4901-F602 (V4-2188) inboard isolation valve
4. 120-V ac					
30 A fuse block (single pole)	Bussmann (FRN)	72B-3A-1A(R)	6.25 A	N.A.	T4700-C001 drywell cooling fan 1 motor winding heater
72B-3A-1A(R) (fuse box R1600S002E)	Bussmann (FRN)	72B-3A-1A(R)	6.25 A	N.A.	T4700-C002 drywell cooling fan 1 motor winding heater
30 A fuse block (single pole)	Bussmann (FRN)	72C-3A-9C	6.25 A	N.A.	T4700-C002 drywell cooling fan 2 motor starter control circuit
72C-3A-9C (fuse box R1600S003H)	Bussmann (FRN)	72C-3A-9C	6.25 A	N.A.	T4700-C002 drywell cooling fan 2 motor starter control circuit
30 A fuse block (single pole)	Bussmann (FRN)	72C-3A-9C	6.25 A	N.A.	T4700-C002 drywell cooling fan 2 motor winding heater
72C-3A-9C (fuse box R1600S003H)	Bussmann (FRN)	72E-5A-1A	6.25 A	N.A.	T4700-C002 drywell cooling fan 2 motor winding heater

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
30 A fuse block (single pole)	Bussmann (FRN)	72E-5A-1A	6.25 A	N.A.	T4700-C003 drywell cooling fan 3 motor winding heater
72E-5A-1A (fuse box R1600S004E)	Bussmann (FRN)	72E-5A-1A	6.25 A	N.A.	T4700-C003 drywell cooling fan 3 motor winding heater
30 A fuse block (single pole)	Bussmann (FRN)	72F-4A-2A(R)	6.25 A	N.A.	T4700-C004 drywell cooling fan 4 motor winding heater
72F-4A(R) (fuse box R1600S005G)	Bussmann (FRN)	72F-4A-2A(R)	6.25 A	N.A.	T4700-C004 drywell cooling fan 4 motor winding heater
5. 130V dc					
4.16kV switchgear Bus 65G, Position G5	Bussmann (FRN)	130v dc at swgr. Bus 65G	15 A (3 fuses)	N.A.	B3100-S001B MG Set B drive motor control
480 V switchgear Bus 72F, Position 2C	Bussmann (FRN)	130V dc at swgr. Bus 72F	15 A (3 fuses)	N.A.	G3303-C001A reactor water clean-up system recirculating Pump 'A' drive motor control circuit
480V switchgear Bus 73E, Position 2D	Bussmann (FRN)	130V dc at swgr. Bus 72E	15 A (3f fuses)	N.A.	G3303-C001B reactor water clean-up system circulating pump 'B' drive motor control circuit

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.3 The thermal overload protection of each valve shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves inoperable, continuously bypass the inoperable thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.3 The thermal overload protection for the above required valves shall be demonstrated OPERABLE at least once per 18 months and following maintenance on the motor starter by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overloads for the above required valves.

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TABLE 3.8.4.3-1

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
1. B21-F016 (V17-2009)	Nuclear Boiler System
B21-F019 (V17-2010)	Nuclear Boiler System
B21-F021 (V10-2004)	Nuclear Boiler System
B21-F041A (V17-2099)	Nuclear Boiler System
B21-F041B (V17-2100)	Nuclear Boiler System
B21-F041C (V17-2101)	Nuclear Boiler System
B21-F041D (V17-2102)	Nuclear Boiler System
B21-F600 (V10-2010)	Nuclear Boiler System
2. B31-F031A (V8-2003)	Reactor Recirculation System
B31-F031B (V8-2004)	Reactor Recirculating system
3. E11-F003A (V8-2141)	Residual Heat Removal (RHR)
E11-F003B (V8-2142)	RHR
E11-F004A (V8-2099)	RHR
E11-F004B (V8-2102)	RHR
E11-F004C (V8-2101)	RHR
E11-F004D (V8-2100)	RHR
E11-F006A (V8-2095)	RHR
E11-F006B (V8-2098)	RHR
E11-F006C (V8-2097)	RHR
E11-F006D (V8-2096)	RHR
E11-F007A (V8-2133)	RHR
E11-F007B (V8-2134)	RHR
E11-F008 (V8-2092)	RHR
E11-F009 (V8-2091)	RHR
E11-F010 (V8-2187)	RHR
E11-F015A (V8-2161)	RHR
E11-F015B (V8-2162)	RHR
E11-F016A (V8-2167)	RHR
E11-F016B (V8-2168)	RHR
E11-F017A (V8-2159)	RHR
E11-F017B (V8-2160)	RHR
E11-F021A (V8-2169)	RHR
E11-F021B (V8-2170)	RHR
E11-F022 (V8-2172)	RHR
E11-F023 (V8-2171)	RHR
E11-F024A (V8-2135)	RHR
E11-F024B (V8-2136)	RHR
E11-F026 (V8-2152)	RHR
E11-F027A (V8-2157)	RHR
E11-F027B (V8-2158)	RHR
E11-F028A (V8-2155)	RHR
E11-F028B (V8-2156)	RHR
E11-F047A (V8-2137)	RHR
E11-F047B (V8-2138)	RHR
E11-F048A (V8-2139)	RHR

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TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
	E11-F048B (V8-2140)	RHR
	E11-F068A (V15-2018)	RHR
	E11-F068B (V15-2019)	RHR
	E11-F073 (V15-2015)	RHR
	E11-F075 (V15-2016)	RHR
	E11-F608 (V8-2407)	RHR
	E11-F611A (V8-4613)	RHR
	E11-F611B (V8-4614)	RHR
4.	E1150-F601A (V15-2127)	RHR Service Water Pumps and Motors
	E1150-F601B (V15-2125)	RHR Service Water Pumps and Motors
	E1150-F602A (V15-2128)	RHR Service Water Pumps and Motors
	E1150-F602B (V15-2126)	RHR Service Water Pumps and Motors
	E1150-F603A (V15-2108)	RHR Service Water Pumps and Motors
	E1150-F603B (V15-2083)	RHR Service Water Pumps and Motors
	E1150-F604A (V15-2109)	RHR Service Water Pumps and Motors
	E1150-F604B (V15-2084)	RHR Service Water Pumps and Motors
	E1150-F605A (V15-2110)	RHR Service Water Pumps and Motors
	E1150-F605B (V15-2085)	RHR Service Water Pumps and Motors
5.	E21-F004A (V8-2019)	Core Spray System
	E21-F004B (V8-2020)	Core Spray System
	E21-F005A (V8-2021)	Core Spray System
	E21-F005B (V8-2022)	Core Spray System
	E21-F015A (V8-2033)	Core Spray System
	E21-F015B (V8-2034)	Core Spray System
	E21-F031A (V8-2031)	Core Spray System
	E21-F031B (V8-2032)	Core Spray System
	E21-F036A (V8-2007)	Core Spray System
	E21-F036B (V8-2008)	Core Spray System
6.	E41-F001 (V17-2022)	High Pressure Coolant Injection System (HPCI)
	E41-F002 (V17-2020)	HPCI
	E41-F003 (V17-2021)	HPCI
	E41-F004 (V8-2191)	HPCI
	E41-F006 (V8-2194)	HPCI
	E41-F007 (V8-2193)	HPCI
	E41-F008 (V8-2198)	HPCI
	E41-F011 (V8-2200)	HPCI
	E41-F012 (V8-2196)	HPCI
	E41-F021 (V11-2006)	HPCI

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
E41-F022 (V11-2008)	HPCI
E41-F041 (V8-2204)	HPCI
E41-F042 (V8-2202)	HPCI
E41-F059 (V8-2218)	HPCI
E41-F075 (V11-2013)	HPCI
E41-F079 (V11-2019)	HPCI
E41-F600 (V17-2088)	HPCI
7. E51-F001 (V11-2002)	Reactor Core Isolation Cooling System (RCIC)
E51-F002 (V8-2235)	RCIC
E51-F007 (V17-2030)	RCIC
E51-F008 (V8-2031)	RCIC
E51-F010 (V8-2221)	RCIC
E51-F012 (V8-2227)	RCIC
E51-F013 (V8-2228)	RCIC
E51-F019 (V8-2230)	RCIC
E51-F022 (V8-2232)	RCIC
E51-F029 (V8-2223)	RCIC
E51-F031 (V8-2225)	RCIC
E51-F045 (V17-2032)	RCIC
E51-F046 (V17-2239)	RCIC
E51-F059 (V17-2023)	RCIC
E51-F062 (V11-2020)	RCIC
E51-F084 (V11-2026)	RCIC
8. G1154-F018 (V9-2022)	Drywell Floor Drain System
G1154-F600 (V9-2044)	Drywell Floor Drain System
9. G33-F001 (V8-2252)	Reactor Water Clean-Up System (RWCU)
G33-F004 (V8-2253)	RWCU
10. G51-F600 (V8-3832)	Torus Water Management System (TWMS)
G51-F601 (V8-3834)	TWMS
G51-F602 (V8-3831)	TWMS
G51-F603 (V8-3833)	TWMS
G51-F604 (V8-3849)	TWMS
G51-F605 (V8-3847)	TWMS
G51-F606 (V8-3850)	TWMS
G51-F607 (V8-3848)	TWMS

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
11. P11-F616 (V8-2790)	Condensate Storage and Transfer System
12. P44-F601A (V8-2323)	Emergency Equipment Cooling Water (EECW)
P44-F601B (V8-2314)	EECW
P44-F602A (V8-2325)	EECW
P44-F602B (V8-2322)	EECW
P44-F603A (V8-2324)	EECW
P44-F603B (V8-2315)	EECW
P44-F604 (V8-2425)	EECW
P44-F605A (V8-2427)	EECW
P44-F605B (V8-2426)	EECW
P44-F606A (V8-2486)	EECW
P44-F606B (V8-2484)	EECW
P44-F607A (V8-2485)	EECW
P44-F607B (V8-2483)	EECW
P44-F608 (V8-2487)	EECW
P44-F613 (V8-3057)	EECW
P44-F614 (V8-3058)	EECW
P44-F615 (V8-3889)	EECW
P44-F616 (V8-3890)	EECW
13. P50-F603 (V5-2006)	Compressed Air Systems
P50-F604 (V5-2007)	Compressed Air Systems
14. T48-F601A (V4-2140)	Containment Atmosphere Control System
T48-F601B (V4-2139)	Containment Atmosphere Control System
T48-F602A (V4-2142)	Containment Atmosphere Control System
T48-F602B (V4-2141)	Containment Atmosphere Control System
T48-F603A (V4-2144)	Containment Atmosphere Control System
T48-F603B (V4-2143)	Containment Atmosphere Control System
T48-F604A (V4-2148)	Containment Atmosphere Control System
T48-F604B (V4-2149)	Containment Atmosphere Control System
T48-F605A (V4-2154)	Containment Atmosphere Control System
T48-F605B (V4-2153)	Containment Atmosphere Control System
T58-F606A (V4-2156)	Containment Atmosphere Control System
T48-F606B (V4-2155)	Containment Atmosphere Control System
T4803-F601 (VR3-3011)	Containment Atmosphere Purging System
T4803-F602 (VR3-3024)	Containment Atmosphere Purging System
15. T49-F601 (V4-2080)	Primary Containment Pneumatic Supply System
T49-F602 (V4-2188)	Primary Containment Pneumatic Supply System

REACTOR PROTECTION SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring assemblies for each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring assembly 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 assemblies for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring assembly to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS electric power monitoring assemblies shall be determined OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed within the previous 6 months.

b. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency 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. Overvoltage $\leq$ 132 VAC,

   2. Undervoltage $\geq$ 108 VAC,

   3. Underfrequency $\geq$ 57 Hz.

STANDBY LIQUID CONTROL SYSTEM ASSOCIATED ISOLATION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.5 All circuit breakers shown in Table 3.8.4.5-1 shall be OPERABLE.

APPLICABILITY: When standby liquid control system (SLCS) is required to be OPERABLE.

ACTION:

When one or more of the circuit breakers shown in Table 3.8.4.5-1 inoperable either:

a. Restore the inoperable circuit breaker(s) to OPERABLE status within 8 hours, or

b. Trip the inoperable circuit breaker(s), rack out or remove the device from service within 8 hours and verify the circuit breaker(s) to be racked out or removed from service at least once per 7 days thereafter, and declare the SLCS inoperable and apply the appropriate ACTION statement for the SLCS.

SURVEILLANCE REQUIREMENTS

4.8.4.5 Each of the above required circuit breaker(s) shall be demonstrated OPERABLE:

a. At least once per 18 months by performing a CHANNEL CALIBRATION of the associated protective relays and a CHANNEL FUNCTIONAL TEST of each breaker which includes simulation of actuation of the system and verifying that each relay and associated circuit breaker and overcurrent control circuits functions as designed.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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TABLE 3.8.4.5-1

STANDBY LIQUID CONTROL SYSTEM ASSOCIATED ISOLATION DEVICES
480 V MOTOR CONTROL CENTERS

MCC 72B-4A	
Position 2F	SLC Heat Trace A
MCC 72B-4C	
Position 1A	Incoming Feed
Position 2AR	SLC Pump A
MCC 72C-4A	
Position 5C	SLC Heater A
MCC 72E-4B	
Position 1AR	SLC Heat Trace B
MCC 72E-5B	
Position 1C	Incoming Line
Position 2B	SLC Pump B
Position 2CR	SLC Heater B

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3/4.9 REFUELING OPERATIONS

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 hoists/grapple fuel-loaded.
4. Fuel grapple position.
5. Service platform 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.

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.

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. The "shorting links" removed from the RPS circuitry prior to and during the time any control rod is withdrawn# and shutdown margin demonstrations are in progress.

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.

d. Verifying, within 8 hours prior to and at least once per 12 hours during, that the RPS circuitry "shorting links" have been removed during:

   1. The time any control rod is withdrawn,** or

   2. Shutdown margin demonstrations.

*Provided signal-to-noise ratio is $\geq$ 2. Otherwise, 3 cps.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

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, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

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 control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

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 of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling platform personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of control rods with their normal drive system.

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling platform crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days prior to the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff when the load exceeds 1200 pounds for the fuel grapple hoist and 1050 pounds for all other cranes or hoists.

b. Demonstrating operation of the uptravel stop when fuel grapple hoist uptravel and frame mounted and monorail auxiliary hoists uptravel reaches 6 feet 6 inches below the top of the platform tracks.

c. Demonstrating operation of the downtravel cutoff when fuel grapple hoist downtravel reaches 52 feet 3 inches below the top of the platform tracks and when frame mounted and monorail auxiliary hoists reach 85 feet below the hoist.

d. Demonstrating operation of the slack cable cutoff when the load is less than 50 pounds for the fuel grapple hoist.

e. Demonstrating operation of the loaded interlock when the load exceeds 535 pounds for the fuel grapple hoist and 450 pounds for all other cranes and hoists.

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3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1100 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Loads, other than fuel assemblies, shall be verified to be less than or equal to 1100 pounds prior to movement over fuel assemblies in the spent fuel storage pool racks.

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3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 20 feet 6 inches 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.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 22 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

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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 CONDITIONS 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.

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.

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.

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.

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.

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.

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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. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 20 feet 6 inches above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within 1 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 SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within 1 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 to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

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 loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 20 feet 6 inches 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 1 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 in operation, within 1 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 to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment 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 to be within the limits at least once per hour during low power PHYSICS TESTS.

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3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that the rod worth minimizer is OPERABLE per Specifications 3.1.4.1:

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 RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RSCS are bypassed, verify:

a. Within 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed:

   1. That the rod worth minimizer is OPERABLE per Specification 3.1.4.1.

   2. That movement of the control rods from 50% ROD DENSITY to the RSCS preset power level is blocked or limited to the single notch mode.

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3, and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1, or 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 "rod-out-notch-override" 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,

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 or 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 pump speed 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 to be less than 24 hours at least once per hour during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

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3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.2 may be suspended during the performance of the Startup Test Program until 6 months after initial criticality.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The number of months since criticality shall be verified to be less than or equal to 6 months at least once per 31 days during the Startup Test Program.

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3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 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

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.3-1) 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, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1.1.1-1.

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. Waste Sample Tanks(3)[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			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 4.11.1.1.1-1 (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 an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

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TABLE 4.11.1.1.1-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This 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 Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[d]A composite sample is one in which the quantity of liquid samples 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. This may be accomplished through composites of grab samples obtained prior to discharge after the tanks have been recirculated.

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, from each reactor unit, 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. This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Safe Drinking Water Act.*

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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*Applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge.

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LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE and appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent, from each reactor unit, 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 any 31-day period.

APPLICABILITY. At all times.

ACTION:

a. With radioactive liquid waste being discharged and in excess of the above limits and any portion of the liquid radwaste treatment system not in operation, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

 1. Explanation of why liquid radwaste was being discharged without complete treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

 2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

 3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each reactor unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

4.11.1.3.2 The installed liquid radwaste treatment system shall be demonstrated OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary 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 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, pursuant to Specification 6.9.1.8.

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 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.

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) 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 tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

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, 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.

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. Containment PURGE	P Each PURGE Grab Sample	P Each PURGE P	Principal Gamma Emitters[b] H-3	$1x10^{-4}$ $1x10^{-6}$
B. Reactor Building Exhaust Plenum Standby Gas Treatment System[h]	M[c,d,e] Grab Sample	M[c] M[c]	Principal Gamma Emitters[b] H-3	$1x10^{-4}$ $1x10^{-6}$
C. Radwaste Building Turbine Building Service Building On-site Storage Facility	M Grab Sample	M M	Principal Gamma Emitters[b] H-3	$1x10^{-4}$ $1x10^{-6}$
D. All Release Types as listed in B and C above.	Continuous[f]	W[g] Adsorbant Sample	I-131	$1x10^{-12}$
	Continuous[f]	W[g] Particulate Sample	Principal Gamma Emitters[b]	$1x10^{-11}$
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$

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 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.

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TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATIONS

[b]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[c]Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period.

[d]Tritium grab samples shall be taken at least once per 24 hours when either the reactor well or the dry-separator storage pool is flooded.

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

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

[g]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

[h]Required when the SGTS is in operation.

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LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 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 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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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, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 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 of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

OFF-GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The OFF-GAS TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser steam jet air ejectors are in operation.

ACTION:

a. With the OFF-GAS TREATMENT 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 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.

c. The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 The OFF-GAS TREATMENT SYSTEM shall be demonstrated OPERABLE by meeting Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

VENTILATION EXHAUST TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM as described in the ODCM shall be OPERABLE and appropriate portions of the system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.3 mrem to any organ in any 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged in excess of the above limits and any portion of the VENTILATION EXHAUST TREATMENT SYSTEM not in operation, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

   1. 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

4.11.2.5.1 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.

4.11.2.5.2 The VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by meeting Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

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 condensor steam jet air ejectors are 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 the waste gases in the main condenser offgas treatment system with the hydrogen monitors required OPERABLE by Table 3.3.7.12-1 of Specification 3.3.7.12.

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MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The gross radioactivity rate of noble gases measured near the main condenser steam jet air ejectors shall be limited to less than or equal to 340 millicuries/sec.

APPLICABILITY: At all times.

ACTION:

With the gross radioactivity rate of noble gases at the main condenser steam jet air ejector exceeding 340 millicuries/sec, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases near the outlet of the main condenser steam jet air ejector shall be continuously monitored in accordance with Specification 3.3.7.12.

4.11.2.7.2 The gross radioactivity rate of noble gases from the main condenser steam jet air ejector shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken near the discharge of the main condenser steam jet air ejector:

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Offgas Radiation 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.

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LIMITING CONDITION FOR OPERATION

3.11.2.8 VENTING or PURGING of the Mark I containment shall be through the standby gas treatment system or the reactor building ventilation system.

APPLICABILITY: Whenever the containment is vented or purged.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING and PURGING of the drywell.

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

SURVEILLANCE REQUIREMENTS

4.11.2.8.1 The containment shall be determined to be aligned for VENTING or PURGING through the standby gas treatment system or the reactor building ventilation system within 4 hours prior to start of and at least once per 12 hours during VENTING or PURGING of the containment.

4.11.2.8.2 Prior to use of the purge system through the standby gas treatment system assure that:

a. Both standby gas treatment system trains are OPERABLE whenever the purge system is in use, and

b. Whenever the purge system is in use during OPERATIONAL CONDITION 1 or 2 or 3, only one of the standby gas treatment system trains may be used.

4.11.2.8.3 The containment drywell shall be sampled and analyzed per Table 4.11-2 of Specification 3.11.2.1 within 8 hours prior to the start of and at least once per 12 hours during VENTING and PURGING of the drywell through other than the standby gas treatment system.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be SOLIDIFIED or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet the requirements for transportation to the disposal site and for receipt at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the solid waste system as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) test the improperly processed waste in each container to ensure that it meets the requirements for transportation to the disposal site and for receipt at the disposal site and (2) take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms) shall be verified in accordance with the PROCESS CONTROL PROGRAM.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 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.

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared out of service, restore the equipment to operable status or provide for contract capability to process wastes as necessary to meet the requirements for transportation to the disposal site and for receipt at the disposal site.

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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 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., 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 includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4, ACTION a.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

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 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12.1-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

*The methodology used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 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.

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]	37 routine monitoring stations, DRI-DR37, with two or more dosimeters placed as follows: 1) an inner ring of stations in the general area of the SITE BOUNDARY and additional rings at approximately 2, 5, and 10 miles, with a station in at least every other meterological sector for each ring with the exception of those sectors over Lake Erie. The balance of the stations, 8, should placed in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	Quarterly	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations, A1-A5: a. 3 samples, A1-A3, from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground-level D/Q. b. 1 sample, A4, from the vicinity of a community having the highest calculated annual average ground-level 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 filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.

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. 1 sample, A5, from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]		
3. WATERBORNE			
a. Surface[f]	a. 1 sample upstream, Wa1 b. 1 sample downstream, Wa2	Composite sample over 1-month period[g]	Gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources, Wb1, Wb2, only if likely to be affected[h].	Quarterly	Gamma isotopic[e] and tritium analysis quarterly if ground water flow reversal is noted.
c. Drinking	a. 1 sample of each of 1 to 3, Wc1-Wc3, of the nearest water supplies that could be affected by its discharge. b. 1 sample from a control location, Wc4.	Composite sample over 2-week period[g] when I-131 analysis is performed, monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[f] Composite for gross beta and gamma isotopic analyses[e] monthly. Composite for tritium analysis quarterly.
d. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value, Wd1.	Semiannually	Gamma isotopic analysis[e] semiannually.

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
4. INGESTION			
a. Milk	a. Samples from milking animals in 3 locations, Ia1-Ia3, within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas, Ia1-Ia3, between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr[f].	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.
	b. 1 sample from milking animals at a control location, Ia4, 15-30 km distant and in the least prevalent wind direction.		
b. Fish and Invertebrates	a. 1 sample of each commercially and recreationally important species in vicinity of plant discharge area, Ib1-Ib_.	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis[e] on edible portions.
	b. 1 sample of same species in areas not influenced by plant discharge, Ib10-Ib__.		
c. Food Products	a. 1 sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged, Ic1-Ic__.	At time of harvest[j]	Gamma isotopic analyses[e] on edible portion.

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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. Food Products (cont'd)	b. Samples of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed, Ic10-Ic__.	Monthly when available	Gamma isotopic[e] and I-131 analysis.
	c. 1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed, Ic20-Ic23.	Monthly when available	Gamma isotopic[e] and I-131 analysis.

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TABLE 3.12.1-1 (Continued)

TABLE NOTATIONS

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, 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. Pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

[c]The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that have valid background data may be substituted.

[d]Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

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TABLE 3.12.1-1 (Continued)

TABLE NOTATION

[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]A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

[h]Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[i]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[j]If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR Part 141 value.

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 Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

TABLE 4.12.1-1 (Continued)

TABLE NOTATIONS

[a] This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

[b] Required detection capabilities for thermoluminescent dosimeters used for environmental measurements 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 valves of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12.1-1 (Continued)

TABLE NOTATIONS

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

[d]LLD for drinking water samples.

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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 a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, visual survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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*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 in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12.1-1, Part 4.c., 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 supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.12.3 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 Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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 3.6.5.3 requires two standby gas treatment 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, within one hour measures must be initiated 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. As a further example, Specification 3.6.6.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

3.0.4 This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS or other 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 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that surveillance activities associated with a Limiting Conditions for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies of performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL CONDITION or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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 insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the 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 anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

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3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than 1.06 during the limiting power transient analyzed in Section 15B 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 systematic 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 compliace 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 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15B.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

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3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides 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. To allow for potential leakage and improper mixing this concentration is increased by 25%. The required concentration is achieved by having a minimum available quantity of 4587 gallons of sodium pentaborate solution containing a minimum of 5500 lbs of sodium pentaborate. This quantity of solution is a net amount which is above the pump suction, thus allowing for the portion which cannot be injected. The pumping rate of 41.2 gpm 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 during shutdown. The temperature 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.

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, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

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1. C. J. Paone, R. C. Stirn and J. A. 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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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 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 and 3.2.1-3.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. <u>Input Changes</u>

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

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AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incorporate 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-upscale scram setting and flow biased simulated thermal power-upscale control rod block functions 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 MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

BASES TABLE B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER.................... 3430 MWt* which corresponds to 105% of rated steam flow

Vessel Steam Output.................. 14.86×10^6 lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure........... 1055 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 4.1 ft^2

b. Small Breaks 0.1 ft

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

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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 transients analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR of 1.06, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figure 3.2.3-1.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15B.0-1 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 nonpressurization 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 K_f factor of Figure 3.2.3-2 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

The K_f factors values shown in Figure 3.2.3-2 were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube setpoint and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated thermal flow.

The K_f factors shown in Figure 3.2.3-2 are conservative for the General Electric plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

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.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the 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 trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 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 a 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 ATWS RECIRCULATION PUMP TRIP SYSTEM ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Appendix 15B.8 of the FSAR.

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.

3/4.3.5 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, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

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 continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, 64.

MONITORING INSTRUMENTATION (Continued)

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and 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 Criterion 19 of 10 CFR Part 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 monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975, 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 of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions 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 they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.8 CHLORINE DETECTION SYSTEM

The OPERABILITY of the chlorine detection system ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine mode of operation to provide the required protection. The detection system required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release", Revision 1, January, 1977.

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both 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 is an integral element 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 must be greater.

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.

3/4.3.7.10 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 mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.11 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.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.12 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

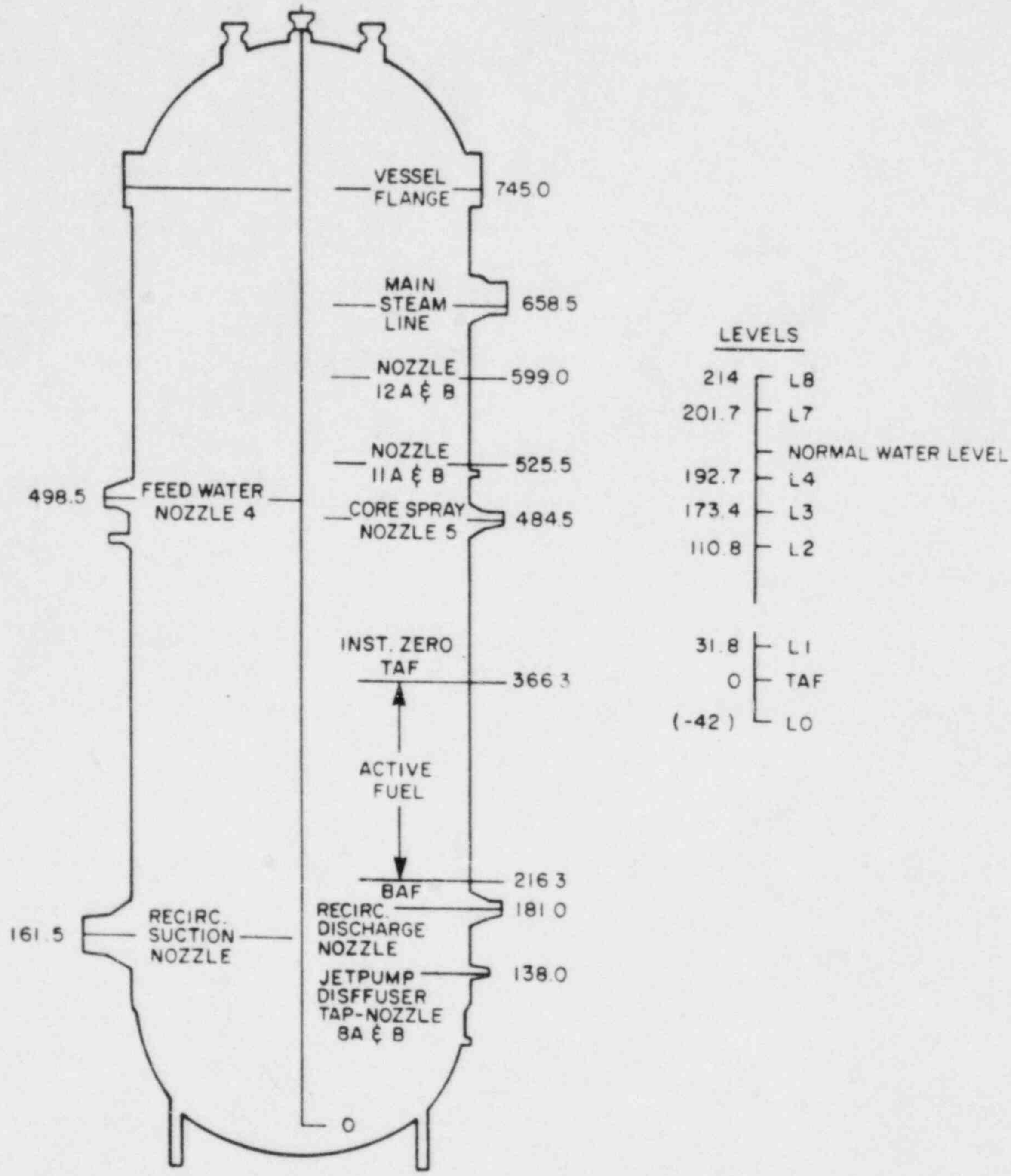
The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM utilizing the system design flow rates as specified in the ODCM. This conservative method is used because the Fermi 2 design does not include flow rate measurement devices. This will ensure the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 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 not required to protect safety-related components, equipment, or structures. However, it is included in order to improve overall plant reliability.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEMS ACTUATION INSTRUMENTATION

The feedwater/main turbine trip systems actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of a high reactor vessel water level due to failure of the feedwater controller under maximum demand.



BASES FIGURE B 3/4.3-1

REACTOR VESSEL WATER LEVEL

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, 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 pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria.

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 was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

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 a potentially high thrust load (designated as load case C.3.3) on the SRV discharge lines is eliminated during subsequent actuations. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. 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.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that certain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

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 system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 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.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 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 phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131 must be restricted to no more than 800 hours per year, approximately 10 percent of the unit's yearly operating time, since these activity levels increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam line rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

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 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases 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 established 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. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, 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.

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PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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 of 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 1974 Edition and Addenda through Summer, 1975.

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 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 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 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

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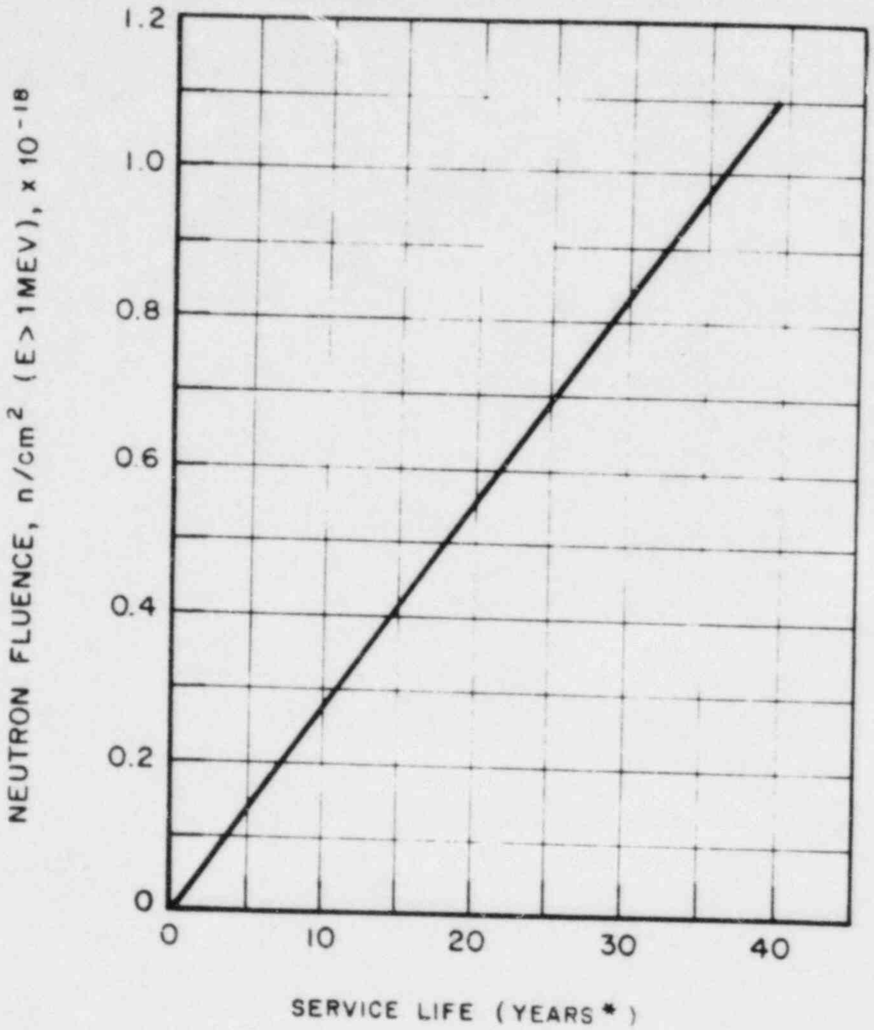

FAST NEUTRON FLUENCE (E>1 MeV) at ¼ T AS A FUNCTION OF SERVICE LIFE*

BASES FIGURE B 3/4.4.6-1

* At 90% of RATED THERMAL POWER and 90% availability.

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3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

The core spray system (CSS), together with the LPCI mode of the RHR system, is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The CSS 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 CSS 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.

If LPCI injection is required when the LPCI system is in the RHR shutdown cooling mode of operation, the motor-operated torus suction valves will require manual operator realignment to facilitate this ECCS operation. All other LPCI components will automatically realign or start as necessary.

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. Two subsystems, each with two pumps, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation 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 high pressure coolant injection (HPCI) 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 HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HCPI system continues to operate until reactor vessel pressure is below the pressure at which CSS system operation or LPCI mode of the RHR system operation maintains core cooling.

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ECCS-OPERATING and SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5000 gpm at differential pressures between 1120 and 150 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 HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI 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 HPCI 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 150 psig. This pressure is substantially below that for which the low pressure cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls five selected safety-relief valves although the safety analysis only takes credit for four valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI, CS and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITIONS 1, 2, or 3 is also required by Specification 3.6.2.1.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

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SUPPRESSION CHAMBER (Continued)

In OPERATIONAL CONDITION 4 and 5 the suppression chamber 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 water volume is based on NPSH, recirculation volume and vortex prevention plus a 2.4' safety margin for conservatism.

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 56.5 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening, and analyzing the Type A test data.

Appendix J to 10 CFR Part 50, Paragraph III.A.3, requires that all Type A tests be conducted in accordance with the provisions of N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." N45.4-1972 requires that Type A test data be analyzed using point-to-point or total time analytical techniques. Specification 4.6.1.2a. requires use of the mass plot analytical technique. The mass plot method is considered the better analytical technique, since it yields a confidence interval which is a small fraction of the calculated leak rate; and the interval decreases as more data sets are added to the calculation. The total time and point-to-point techniques may give confidence intervals, which are large fractions of the calculated leak rate, and the intervals may increase as more data sets are added.

The mass plot method is endorsed by ANSI/ANS 56.8-1981 (Containment System Leakage Requirements) which superseded N45.4-1972.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be

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PRIMARY CONTAINMENT AIR LOCKS (Continued)

long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 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 56.5 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 56.5 psig does not exceed the maximum allowable pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The 20- and 24-inch drywell and suppression chamber 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. Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 20- and 24-inch valves cannot be inadvertently opened,

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

they are sealed closed in accordance with Standard Review Plan 6.2.4, which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.

The use of the drywell and suppression chamber purge lines is restricted to the 6-inch purge supply and exhaust isolation valves since, unlike the 20- and 24-inch valves, the 6-inch valves will close during a LOCA or steam line break accident and therefore the SITE BOUNDARY dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The design of the 6-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

Leakage intergrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

The 6, 10, 20, and 24 inch purge valves are generally configured in a three (3) valve arrangement at each of the associated purge penetrations. The valves are leak tested by pressurizing between the three valves and a total leakage is determined as opposed to a single valve leakage. Verifying that the measured leakage rate is less than 0.5 L_a for this multi-valve arrangement is more conservative than a limit of 0.5 L_a for a single valve.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the maximum allowable pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

DEPRESSURIZATION SYSTEMS (Continued)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is approximately 56.5 psig which is below the maximum allowable pressure of 62 psig. Maximum water volume of 124,220 ft^3 results in a downcomer submergence of 3'4" and the minimum volume of 121,030 ft^3 results in a submergence of 3'0". The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operation conditions, a design basis accident blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 135°F in the short term following the blowdown. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependence on containment overpressure for post-LOCA operations.

The large thermal capacitance of the suppression pool is also utilized during plant transients requiring safety/relief valve (SRV) actuation. Steam is discharged from the main steam lines through the SRVs and their accompanying discharge lines into the suppression pool where it is condensed, resulting in an increase in the temperature of the suppression pool water. Although stable steam condensation is expected at all pool temperatures, NUREG-0783 imposes a local temperature limit shown in Figure B 3/4.6.2-1 in the vicinity of the T-type quencher discharge device. The limiting plant transients with respect to heat input to the suppression pool have been analyzed. The conservative analysis showed that by limiting the average water temperature to less than or equal to 170°F will result in local pool temperatures below the condensation stability limit of Bases Figure B 3/4.6.2-1.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual

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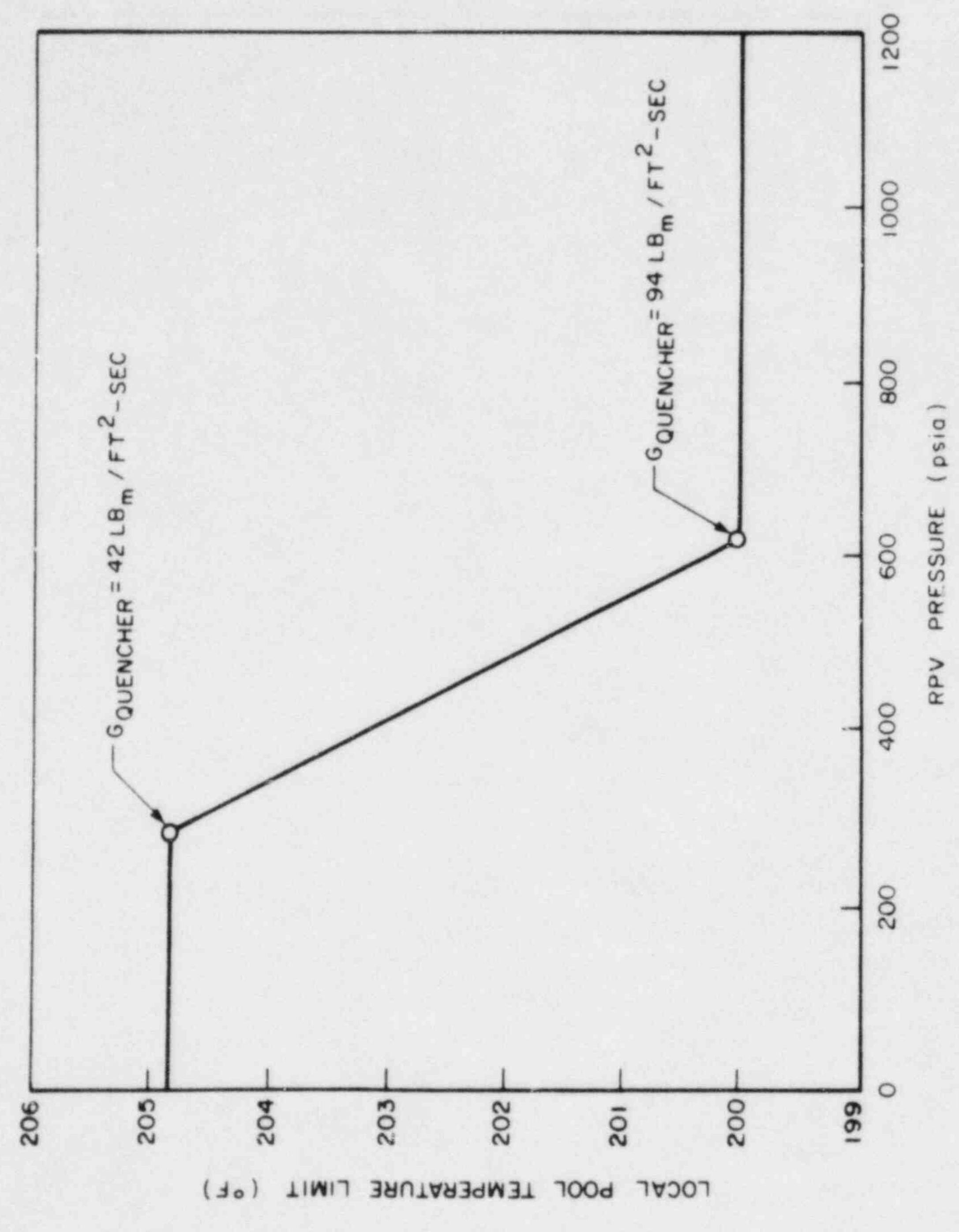

BASES FIGURE B 3/4.6.2-1

LOCAL POOL TEMPERATURE LIMIT

DEPRESSURIZATION SYSTEMS (Continued)

examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety/relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell and between the Reactor Building and suppression chamber. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures. There are valves to provide redundancy so that operation may continue for up to 72 hours with redundant vacuum breakers inoperable in the closed position.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times, the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

SECONDARY CONTAINMENT (Continued)

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continous 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.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. Either drywell and suppression chamber hydrogen recombiner is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

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3/4.7.1 SERVICE WATER SYSTEMS

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

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that (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. Continous 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 SHORE BARRIER PROTECTION

The purpose of the shore barrier is to protect the site backfill from wave erosion.

Category 1 structures are designed to withstand the impact of waves up to 5.4 feet. So long as the backfill is in place, waves greater than 5.4 feet cannot impact Category 1 structures because of the lack of sufficient depth of water to generate such waves.

The shore barrier can sustain a high degree of damage and still perform its function, protecting the site backfill from erosion. Thus the operability condition of the shore barrier has been written to depend upon an annual inspection and survey and an inspection and survey following a severe storm.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 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.

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REACTOR CORE ISOLATION COOLING SYSTEM (Continued)

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCI system and justifies the specified 14-day out-of-service period.

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.

3/4.7.5 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilitizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Onsite Review Organization. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessiblity during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system

SNUBBERS (Continued)

to be unprotected and to result in failure during an assumed initiating event. Inspections performed before the interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.5-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7.5-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

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

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.) The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.6 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 7.7 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers systems, CO_2 systems, Halon systems, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurances that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying 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.

3/4.7.8 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.

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis for FSAR Chapter 15.

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

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

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

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, susbsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator is a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

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

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

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 450-1972, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

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

Table 4.8.2.1-1 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 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 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 with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.020 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

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

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

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

Circuit breakers actuated by fault currents are used as isolation devices to protect equipment associated with the Standby Liquid Control System. The OPERABILITY of these circuit breakers will ensure that the SLCS equipment is protected in the event of faults in the loads powered by these circuit breakers.

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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 the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

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 crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the 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 shutdown cooling mode loop be OPERABLE 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 20 feet 6 inches 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 20 feet 6 inches of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event 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.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize 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

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DOSE (Continued)

ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to their release to the environment. The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the 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.

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3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of 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

DOSE - NOBLE GASES (Continued)

THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

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 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.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

3/4.11.2.4 OFF-GAS TREATMENT SYSTEM

The OPERABILITY of the OFF-GAS TREATMENT SYSTEM ensures that the system 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 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.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the main condenser offgas system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MAIN CONDENSER

Restricting the gross radioactivity rate of 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.

3/4.11.2.8 VENTING OR PURGING

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS.

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3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to 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 only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12.1-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, from visual 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) a vegetation yield of 2 kg/m^2.

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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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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

SITE BOUNDARIES

5.1.3 The site boundaries for gaseous effleunts and for liquid effluents shall be as shown in Figure 5.1.3-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a steel pressure vessel in the shape of a torus. The drywell has a total minimum free air volume of 163,730 cubic feet. The torus has a minimum air volume of 127,760 cubic feet and a minimum water volume of 121,080 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure: 56 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool 281°F.

c. Maximum external pressure: 2 psig.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building and the main steam tunnel and has a minimum free volume of 2,800,000 cubic feet.

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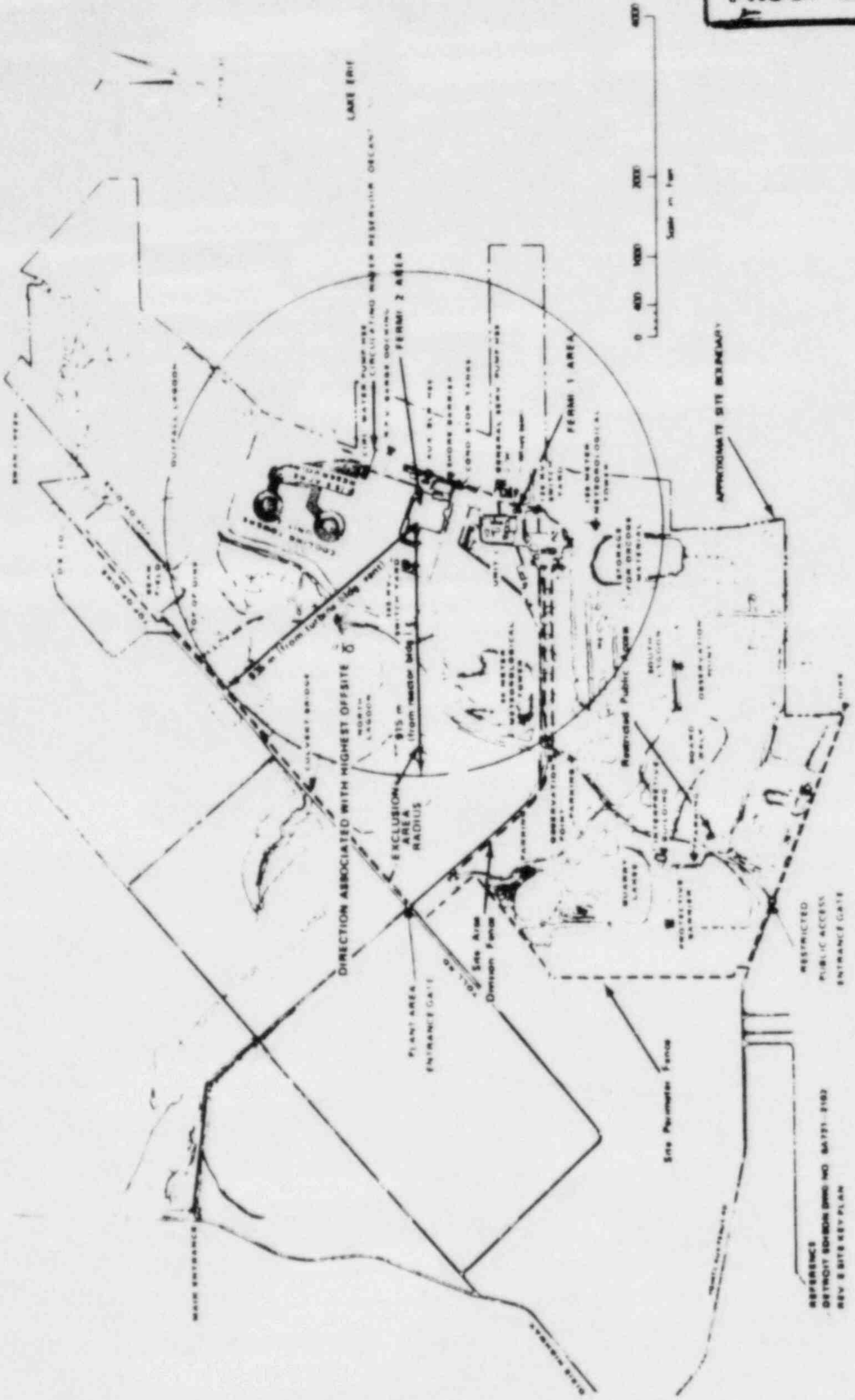
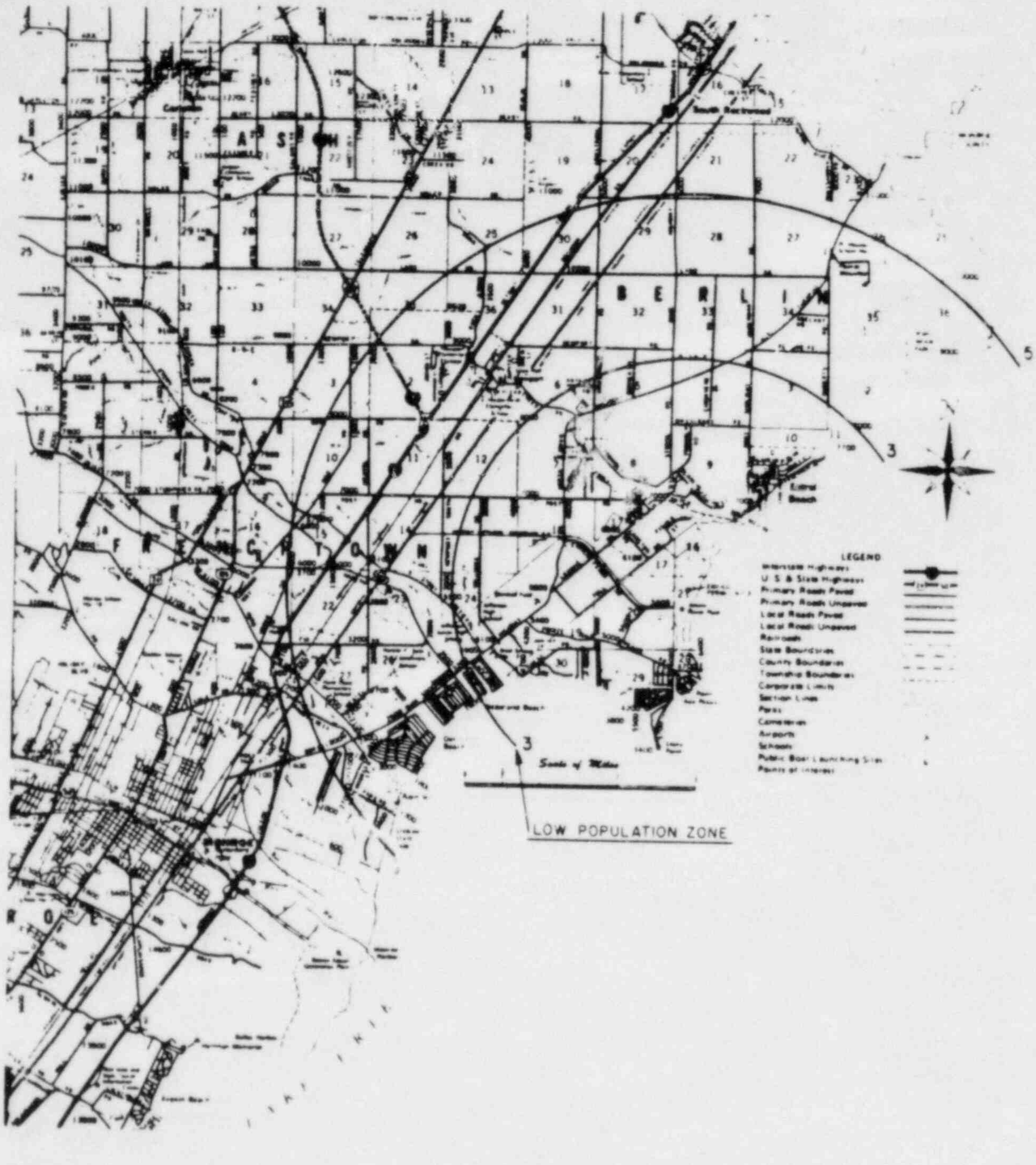


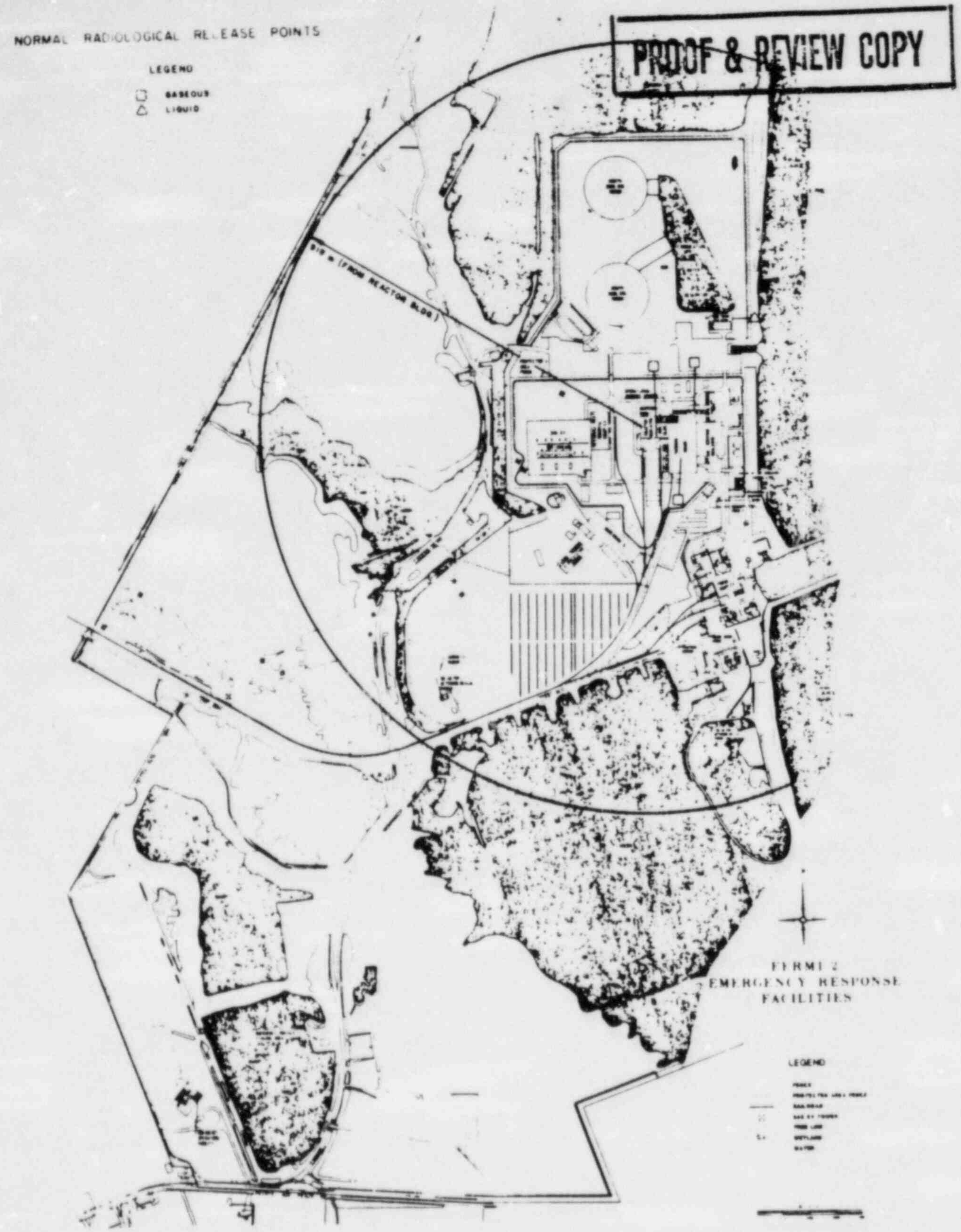
EXCLUSION AREA

FIGURE 5.1.1-1



LOW POPULATION ZONE

FIGURE 5.1.2-1



MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

FIGURE 5.1.3-1

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies with each fuel assembly containing 62 fuel rods and two water rods 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 1.88 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum assembly average enrichment of 3.20 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump.
   2. 1500 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
   3. 1500 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 22,034 cubic feet at a nominal steam dome saturation temperature of 540°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1.9% delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 6.22 and 11.9 x 6.6 inch center-to-center distance between fuel assemblies placed in the high density and low density storage racks, respectively.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 660' 11½".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2383 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

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
	70 step change cycles	Loss of feedwater heaters resulting in a decrease of ≤ 100°F in final feedwater heater outlet temperature
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	130 hydrostatic pressure and leak tests	Pressurized to ≥ 930 and ≤ 1250 psig

SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Superintendent-Nuclear Production shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Nuclear Shift Supervisor (or during his absence from the control room) a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President-Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown 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 Health Physics 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 Nuclear Shift Supervisor, the Shift Technical Advisor, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Health Physics Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, 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 normal 8-hour day, 40-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, on 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 7 day period, all excluding shift turnover time.

3. A break of at least 8 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 Superintendent-Nuclear Production or his deputy, 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 Superintendent-Nuclear Production 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

OFFSITE ORGANIZATION

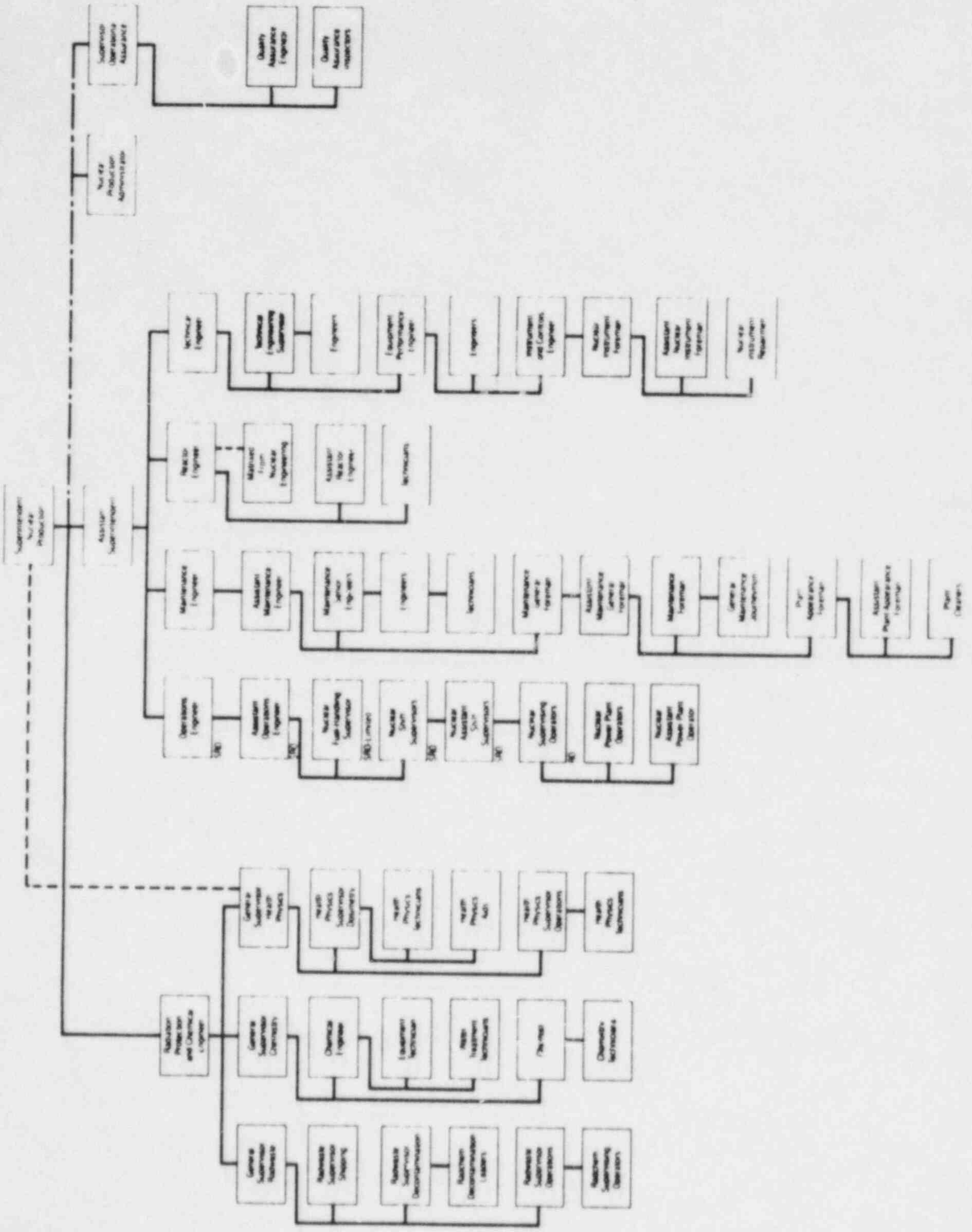

FIGURE 6.2.2-1
UNIT ORGANIZATION

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TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
NSS	1	1
NASS	1	None
NSO	2	1
NPPO/NAPPO	2	1
STA	1	None

TABLE NOTATION

NSS - Nuclear Shift Supervisor with a Senior Operator license
NASS - Nuclear Assistant Shift Supervisor with a Senior Operator license
NSO - Nuclear Supervising Operator with an Operator license
NPPO/NAPPO - Nuclear Power Plant Operator or Nuclear Assistant Power Plant Operator
STA - Shift Technical Advisor

Except for the Nuclear Shift Supervisor, 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 Nuclear Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear 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 ACTIVITIES

FUNCTION

6.2.3.1 Nuclear Safety and Plant Engineering (NSPE) will, as part of its function, have the responsibility for the performance of the onsite Independent Safety Engineering activities. NSPE shall examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The lead Independent Safety Engineer shall submit detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant safety to the Director of Nuclear Engineering.

COMPOSITION

6.2.3.2 NSPE will provide a lead engineer who will be responsible for the performance and accountability of the Independent Safety Engineering activities. He will have available to him the sufficient technical support necessary to effectively conduct these activities. The Director of Nuclear Engineering will provide support equivalent to five engineers, including the lead engineer, to the Independent Safety Engineering activities. Engineers conducting these activities will be located on site, each with a bachelor's degree in Engineering or related science and at least 2 years of professional-level experience in his field.

RESPONSIBILITIES

6.2.3.3 NSPE will, as part of its duties, be responsible for maintaining surveillance of the plant operations and maintenance activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practicable. The lead Independent Safety Engineer is responsible for submitting detailed recommendations to the Director of Nuclear Engineering who is responsible for the followup review and implementation of those recommendations that are warranted and have been approved by the appropriate Nuclear Operations Management and line organizations.

RECORDS

6.2.3.4 Records of Independent Safety Engineering activities performed shall be prepared and maintained by the lead Independent Safety Engineer and a copy forwarded each calendar month to the Director of Nuclear Engineering. In addition, the lead Independent Safety Engineer will issue a monthly summary report of the Independent Safety Engineering activities to the Director of Nuclear Engineering.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Nuclear Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The

*Not responsible for sign-off function.

SHIFT TECHNICAL ADVISOR (Continued)

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.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Supervisor 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, shall meet or exceed the requirements and recommendations of Section 5 of ANSI N18.1-1971 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 Onsite Review Organization (OSRO)

FUNCTION

6.5.1.1 The OSRO shall function to advise the Superintendent-Nuclear Production on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The OSRO shall be composed of the:

Chairman	Superintendent-Nuclear Production
Vice-Chairman	Assistant Superintendent-Nuclear Production
Second Vice-Chairman	Operations Engineer
Secretary	Assistant Operations Engineer
Member	Technical Engineer
Member	Maintenance Engineer
Member	Radiation Protection-Chemical Engineer
Member	Supervisor Operational Assurance
Member	Reactor Engineer
Member	Administrator

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSRO Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSRO activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSRO shall meet at least once per calendar month and as convened by the OSRO Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the OSRO necessary for the performance of the OSRO responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The OSRO shall be responsible for:

a. Review of (1) all proposed procedures required by Specification 6.8 and changes thereto, (2) all proposed programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Superintendent-Nuclear Production to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Manager-Nuclear Operations and to the Nuclear Safety Review Group;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Superintendent-Nuclear Production or the Nuclear Safety Review Group;

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Nuclear Safety Review Group;

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Nuclear Safety Review Group;

RESPONSIBILITIES (Continued)

k. Review of any accidental, unplanned or uncontrolled radioactive release including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Vice President-Nuclear Operations and to the Nuclear Safety Review Group; and

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

6.5.1.7 The OSRO shall:

a. Recommend in writing to the Superintendent-Nuclear Production approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation.

b. Render determinations in writing to the Nuclear Safety Review Group with regard to whether or not each item considered under Specification 6.5.1.6a. through e. constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Manager-Nuclear Operations and the Nuclear Safety Review Group of disagreement between the OSRO and the Superintendent-Nuclear Production; however, the Superintendent-Nuclear Production shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The OSRO shall maintain written minutes of each OSRO meeting that, at a minimum, document the results of all OSRO activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Manager-Nuclear Operations and the Nuclear Safety Review Group.

6.5.2 NUCLEAR SAFETY REVIEW GROUP (NSRG)

FUNCTION

6.5.2.1 The NSRG shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NSRG shall report to and advise the Vice President-Nuclear Operations on those areas of responsibility in Specifications 6.5.2.7 and 6.5.2.8.

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COMPOSITION

6.5.2.2 The Vice-President-Nuclear Operations shall appoint at least nine members to the NSRG and shall designate from this membership a Chairman and at least one Vice Chairman. The membership shall collectively possess experience and competence to provide independent review and audit in the areas listed in Section 6.5.2.1 The Chairman and Vice Chairman shall have nuclear background in engineering or operations and shall be capable of determining when to call in experts to assist the NSRG review of complex problems. All members shall have at least a bachelor's degree in engineering or related sciences. The Chairman shall have at least 10 years of professional level management experience in the power field and each of the other members shall have at least 5 years of cumulative professional level experience in one or more of the fields listed in Section 6.5.2.1 with the exception of the QA member. The QA member shall have at least 2 years of professional level experience in QA and inspection.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NSRG Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSRG activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRG Chairman to provide expert advice to the NSRG.

MEETING FREQUENCY

6.5.2.5 The NSRG shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NSRG necessary for the performance of the NSRG review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four NSRG members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NSRG 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;

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REVIEW (Continued)

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 OSRO.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the NSRG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire unit staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. 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 a qualified offsite licensee fire protection engineer(s) or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least every third year;

AUDITS (Continued)

g. Any other area of unit operation considered appropriate by the NSRG or the Vice President-Nuclear Operations;

h. The radiological environmental monitoring program and the results thereof at least once per 12 months;

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months; and

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

RECORDS

6.5.2.9 Records of NSRG activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRG meeting shall be prepared, approved, and forwarded to the Vice President-Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Vice President-Nuclear Operations within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Vice President-Nuclear Operations and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the OSRO, and the results of this review shall be submitted to the NSRG and the Manager-Nuclear Operations.

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 Manager-Nuclear Operations and the NSRG shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the OSRO. 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 NSRG, and the Manager-Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the Fermi 2 commitments made in response to the requirements of NUREG-0737.

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, using the guidance in Regulatory Guide 1.21 Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

PROCEDURES AND PROGRAMS (Continued)

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the OSRO and shall be approved by the Superintendent-Nuclear Production prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Operator license on Fermi 2; and

c. The change is documented, reviewed by the OSRO, and approved by the Superintendent-Nuclear Production within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the HPCI, CS, RHR, RCIC, reactor water sampling, containment sampling, reactor water cleanup, combustible gas control, control rod drive discharge headers, and standby gas treatment systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

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.

PROCEDURES AND PROGRAMS (Continued)

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. High Density Spent Fuel Racks

A program which will assure that any unanticipated degradation of the high density spent fuel racks will be detected and will not compromise the integrity of the racks.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report 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.

ANNUAL REPORTS (Continued)

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of plant, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket or thermoluminescent dosimeters (TLD) dosimeters 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; and

b. Documentation of all challenges to main steam line safety/relief valves, and

c. A summary of ECCS outage data including:

   1. ECCS outage dates and duration of outages,
   2. Cause of each ECCS outage,
   3. ECCS systems and components in the outage, and
   4. Corrective action taken.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience 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, with operational controls, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.
**A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORTS (Continued)

radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12.1-1; discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.*** This same report shall include an assessment of the

*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

**A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

***In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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 5.1.3-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report 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.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

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

6.10.2 The following records shall be retained for at least 5 years:

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

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

c. ALL REPORTABLE EVENTS.

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

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

f. Records of radioactive shipments.

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

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

6.10.3 The following records shall be retained for the duration of the unit Operating License:

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

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

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

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

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

RECORD RETENTION (Continued)

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.

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 OSRO and the NSRG.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.5 including the date at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/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:

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*Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

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. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrems shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Nuclear Shift Supervisor on duty and/or the health physic supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, 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:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

*Measurement made at 18 inches from source of radioactivity.

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PROCESS CONTROL PROGRAM (PCP) (Continued)

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the OSRO.

b. Shall become effective upon review and acceptance by the OSRO.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the OSRO.

b. Shall become effective upon review and acceptance by the OSRO.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the OSRO. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS (Continued)

2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the OSRO.

b. Shall become effective upon review and acceptance by the OSRO.