JAN 30 1985

Docket No.: 50-458

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

Dear Mr. Cahill:

Subject: River Bend Station Technical Specifications, First Draft

Please find enclosed the typed first draft of the River Bend Station Technical Specifications. Earlier this month you were forwarded the hand marked version of this draft. Members of your staff and the NRC will be meeting at various times in the next few months to finalize these Technical Specifications for attachment to the River Bend Station Operating License.

If you have any questions concerning these specifications or the procedure to be used in their preparation, please contact Licensing Project Manager, Edward Weinkam.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

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

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River Bend Station

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

cc: Troy B. Conner, Jr., Esq.
Conner and Wetterhahn
1747 Pennsylvania Avenue, N. W.
Washington, D.C. 20006

Mr. William J. Reed, Jr.
Director - Nuclear Licensing
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704

H. Anne Plettinger
3456 Villa Rose Dr.
Baton Rouge, Louisiana 70806

Richard M. Troy, Jr., Esq.
Assistant Attorney General in Charge
State of Louisiana Department of Justice
234 Loyola Avenue
New Orleans, Louisiana, 70112

Dwight D. Chamberlain
Resident Inspector
Post Office Box 1051
St. Francisville, Louisiana 70775

Gretchen R. Rothschild
Louisianians for Safe Energy, Inc.
1659 Glenmore Avenue
Baton Rouge, Louisiana 70775

James W. Pierce, Jr., Esq.
P. O. Box 23571
Baton Rouge, Louisiana 70893

Ms. Linda B. Watkins/Mr. Steven Irving
Attorney at Law
355 Napoleon Street
Baton Rouge, Louisiana 70802

Mr. David Zaloudek
Nuclear Energy Division
Louisiana Department of
Environmental Quality
Post Office Box 14690
Baton Rouge, Louisiana 70898

Mr. J. David McNeill, III
William G. Davis, Esq.
Department of Justice
Attorney General's Office
7434 Perkins Road
Baton Rouge, Louisiana 70808

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

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

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

CRITICAL POWER RATIO

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

DOSE EQUIVALENT I-131

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

DRYWELL INTEGRITY

1.11 DRYWELL INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. All drywell equipment hatches are closed and sealed.

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

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

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

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

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

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

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

FRACTION OF LIMITING POWER DENSITY

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM

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

IDENTIFIED LEAKAGE

1.19 IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

1.20 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

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

LINEAR HEAT GENERATION RATE

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

LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MEMBER(S) OF THE PUBLIC

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

MINIMUM CRITICAL POWER RATIO

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

OPERABLE - OPERABILITY

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

OPERATIONAL CONDITION - CONDITION

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PRIMARY CONTAINMENT INTEGRITY

1.31 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

b. All containment equipment hatches are closed and sealed.

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

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

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

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

PROCESS CONTROL PROGRAM (PCP)

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

RATED THERMAL POWER

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

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 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

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

ROD DENSITY

1.36 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

1.37 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

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

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

c. The Standby Gas Treatment System is in compliance with the requirements of Specification 3.6.5.3.

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

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

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

g. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.1.a.

SHUTDOWN MARGIN

1.38 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

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.42 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

1.44 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be that time interval from when the (monitored parameter exceeds its actuation setpoint at the channel sensor) (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

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

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown#,***	> 200°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 200°F
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°F

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

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

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

**See Special Test Exceptions 3.10.1 and 3.10.3.

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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 or equal to 785 psig and core flow greater than or equal to 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

With the reactor coolant system pressure, 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.

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	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	≤ 0.66 W+48%, with a maximum of	≤ 0.66 W+51%, with a maximum of
2) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 8.9 inches above instrument zero*	≥ 8.3 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	≤ 52.0 inches above instrument zero*	≤ 52.6 inches above instrument zero
6. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
7. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
8. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
9. Scram Discharge Volume Water Level - High	≤ 36% of full scale	≤ 39% of full scale
10. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 530 psig	≥ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

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

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.

2.1 SAFETY LIMITS

BASES

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 x 103 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 103 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.

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 1971 Edition, including Addenda through Summer 1973, which permits a maximum pressure transient of 110%, 1375 psig, of design pressure, 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor vessel steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the (USAS Piping Code, Section B31.1), (and the) (ASME Boiler and Pressure Vessel Code, (___) Edition, including Addenda through ______ 19__ for the reactor recirculation piping), which permits a maximum pressure transient of (120)%, (1380) psig, of design pressure, (1150) psig for suction piping and (1250) psig for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the (applicable codes).

2.1.4 REACTOR VESSEL WATER LEVEL

With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect ofdecay 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.

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, and backup to the APRM prior to entering the "RUN" mode.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.1.12 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at 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 criterion of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

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

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-High setpoint; i.e., for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-High setpoint, a time constant of 6 ± 0.6 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

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

3. Reactor Vessel Steam Dome Pressure-High

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Reactor Vessel Water Level-High

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

6. Main Steam Line Isolation Valve-Closure

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

7. Main Steam Line Radiation-High

The main steam line radiation detectors are provided to detect a gross failure of the fuel cladding. When the high radiation is detected, a trip is initiated to reduce the continued failure of fuel cladding. At the same time the main steam line isolation valves are closed to limit the release of fission products. The trip setting is high enough above background radiation levels to prevent spurious trips yet low enough to promptly detect gross failures in the fuel cladding.

8. Drywell Pressure-High

High pressure in the drywell could indicate a break in the primary pressure boundary systems. The reactor is tripped in order to minimize the possibility

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

of fuel damage and reduce the amount of energy being added to the coolant and to the primary containment. The trip setting was selected as low as possible without causing spurious trips.

9. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of 26 gallons of water.

10. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 5% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient.

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

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection with or without coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than 20 milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two twice logic input to the Reactor Protection System. This trip setting, a slower closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section (15.2.2) of the Final Safety Analysis Report.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

12. 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.

13. Manual Scram

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

SECTIONS 3.0 and 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 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.

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:

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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.

*Except movement of IRMs, SRMs or special movable detectors.

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.

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

   1. Within one hour:

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

      b) Disarm the associated directional control valves** either:

         1) Electrically, or

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

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

   2. Comply with Surveillance Requirement 4.1.1.c within 12 hours.

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

a) Electrically, or

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

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

a) Electrically, or

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

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

3. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

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

4.1.3.1.2 When above the low power setpoint of the RPCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

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

*May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.
**These valves may be closed intermittently for testing under administrative controls.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

   2. Open when the scram signal is reset.

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

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

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

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

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

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

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

LIMITING CONDITION FOR OPERATION Continued

ACTION: (Continued)

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

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

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

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

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

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

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.

*Except movement of SRM, IRM, or special removable detectors or normal control rod movement.

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

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

b) OPERABLE.

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

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

d. The provisions of Specification 3.0.4 are not applicable.

3.1.3.3 All control rod scram accumulators shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable, within 8 hours:

a) Restore the inoperable accumulator to OPERABLE status, or

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

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

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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

2. With more than one control rod scram acculululator inoperable the associated control rods inoperable and:

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

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

      1) Electrically, or

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

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

b. In OPERATIONAL CONDITION 5*:

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

      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 accumlator inoperable or with no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

c. The provisions of Specification 3.0.4 are not applicable.

*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

4.1.3.3 Each control rod scram accumulator shall be determined OPERABLE:

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

b. At least once per 18 months by:

1. Performance of a:

a) CHANNEL FUNCTIONAL TEST of the leak detectors, and

b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of 1520 + 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, starting at normal system operating pressure, with no control rod drive pump operating.

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

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

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

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

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

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

      a) Electrically, or

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

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

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

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

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

CONTROL ROD DRIVE COUPLING

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

   a) Electrically, or

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD POSITION INDICATION

LIMITING CONDITION FOR OPERATION

3.1.3.5 At least one control rod position indication system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

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

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

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

   3. When THERMAL POWER is:

      a) Within the low power setpoint of the RPCS:

         1) Declare the control rod inoperable, and

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

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

         1) Electrically (by bypassing on the RGDS analyzer card), or

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

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

b. In OPERATIONAL CONDITION 5* with (a) (both position indicators of 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.

---

*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

4.1.3.5 The above required 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 4.1.3.4.b, and

d. When the alternate control rod position indicator is OPERABLE, by performance of a CHANNEL CHECK at least once per 12 hours.

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

CONTROL ROD WITHDRAWAL

LIMITING CONDITION FOR OPERATION

3.1.4.1 Control rods shall not be withdrawn.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

REACTIVITY CONTROL SYSTEMS

ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#.

ACTION:

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

   1. THERMAL POWER less than or equal to the Low Power Setpoint, control rod movement shall not be permitted, except by a scram.

   2. THERMAL POWER greater than the Low Power Setpoint, control rod withdrawal shall not be permitted.

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

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

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

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

         1) Electrically, or

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

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

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

*See Special Test Exception 3.10.2

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

ACTION (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

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

   2. Prior to other control rod movement after the rod inhibit mode is automatically initiated at the RPCS low power setpoint, (20 + 15, - 0)% of RATED THERMAL POWER, during power reduction.

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

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

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

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

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 one 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 in the storage tank 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 for the percent weight concentration determined once per 31 days per Specification 4.1.5.6.2.

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

b. At least once per 31 days by;

1. Verifying the continuity of the explosive charge.

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

SURVEILLANCE REQUIREMENTS (Continued)

2. Determining that the available weight of sodium pentaborate is greater than or equal to 4246 lbs and the percent weight concentration of sodium pentaborate in solution is within the limits of Figure 3.1.5-1 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 per pump at a pressure of greater than or equal to 1220 psig is met.

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

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

2. Demonstrating that 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:

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

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

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 70°F.

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

SODIUM PENTABORATE SOLUTION
TEMPERATURE/CONCENTRATION REQUIREMENTS

Figure 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 or 3.2.1-5, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

POWER DISTRIBUTION LIMITS

Figure 3.2.1-1

Figure 3.2.1-2

Figure 3.2.1-3

Figure 3.2.1-4

Figure 3.2.1-5

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-high scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

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

POWER DISTRIBUTION LIMITS

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With MCPR less than the applicable MCPR limit shown in Figures 3.2.3-1 and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than (25)% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

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

Core Flow, % of Rated Core Flow

$MCPR_f$

Figure 3.2.3-1

THERMAL POWER, % of RATED THERMAL POWER

$MCPR_p$

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.

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 one hour. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
9. Scram Discharge Volume Water Level - High	1, 2	3	1
	$5^{(g)}$	3	3
10. Turbine Stop Valve - Closure	$1^{(h)}$	4	11
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(h)}$	2	6
12. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
13. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

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 one hour.

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

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

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

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

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

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

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

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

ACTION 11 - Within one hour, place the inoperable instrument channels in both trip systems in the tripped condition and initiate a reduction in THERMAL POWER within 15 minutes and reduce the turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

*Except movement of IRM, SRM or special moveable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

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TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

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

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control is withdrawn* and shutdown margin demonstrations are being performed per Specification 3.10.3.

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

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

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

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

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

(h) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 250 psig,** equivalent to THERMAL POWER less than (40)% of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this etpoint shall be submitted to the Commission within 90 days of test completion.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	$\leq$ 0.09**
c. Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.35
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Reactor Vessel Water Level - High, Level 8	$\leq$ 1.05
6. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
7. Main Steam Line Radiation - High	NA
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	NA
10. Turbine Stop Valve - Closure	$\leq$ 0.06
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$\leq$ 0.07#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Not including simulated thermal power time constant, 6 ± 0.6 seconds.
#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b)	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U,S,(b)	S/U[c], W	SA	2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power - High	S,D[h]	S/U[c], W	W[d][e], SA, R[i]	1
c. Neutron Flux - High	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R[g]	1, 2[j]
4. Reactor Vessel Water Level - Low, Level 3	S	M	R[g]	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R[g]	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2[j]
8. Drywell Pressure - High	S	M	R[g]	1, 2[l]

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Trans/Trip Unit	S	M	R(g)	1, 2, 5(k)
b. Float Switch	NA	Q	R	1, 2, 5(k)
10. Turbine Stop Valve - Closure	S(m)	M	R	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	M	R	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

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

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

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

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 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.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

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

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 pe 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. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low-Low, Level 2	1, 7, 8, 9(b)	2	1, 2, 3 and #	20
b. Drywell Pressure - High	1, 3, 8(b)(c)(k)	2	1, 2, 3	20
c. Containment Purge Isolation Radiation - High	8	1	1, 2, 3 and *	21
d. Manual Initiation	1, 3, 7, 8	2/group	1, 2, 3 and *#	22
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level-Low Low Low, Level 1	6(c)	2	1, 2, 3	20
b. Main Steam Line Radiation - High	6, 9(d)	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	6	2	1	24
d. Main Steam Line Flow - High	6	2/line(e)	1, 2, 3	23
e. Condenser Vacuum - Low	6	2	1, 2**, 3**	23
f. Main Steam Line Tunnel Temperature - High	6	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	6	2	1, 2, 3	23
h. Main Steamline Area Temperature High (Turbine Building)	6	2	1, 2, 3	23
i. Manual Initiation	6	2/group	1, 2, 3	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	11, 12, 13(b)(f)(i)(j)	2	1, 2, 3, and #	25
b. Drywell Pressure - High	11, 12, 13(b)(c)(f)(i)(j)	2	1, 2, 3	25
c. Fuel Handling Area Building Ventilation Exhaust Radiation - High	13(f)(i)	2	1, 2, 3, and *	28
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	12(b)(f)(j)	1	1, 2, 3, and *	29
e. Reactor Building Annulus Pressure Control System - Air Flow Low	12(b)(f)(j)	1	1, 2, 3	29
f. Manual Initiation	11, 12, 13	1/group	1, 2, 3	26
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	7	1	1, 2, 3	27
b. Δ Flow Timer	7	1	1, 2, 3	27
c. Equipment Area Temperature - High	7	1	1, 2, 3	27
d. Equipment Area Δ Temp. - High	7	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	7	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	7	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (continued)				
g. Main Steam Line Tunnel Δ Temp. - High	7	1	1, 2, 3	27
h. SLCS Initiation	7[(g)]	1[(g)]	1, 2, 3	27
i. Manual Initiation	7	1/group	1, 2, 3	26
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	2	1	1, 2, 3	27
b. RCIC Steam Line Flow - High Timer	2	1	1, 2, 3	27
c. RCIC Steam Supply Pressure - Low	2[(h)]	1	1, 2, 3	27
d. RCIC Turbine Exhaust Diaphragm Pressure - High	2	2	1, 2, 3	27
e. RCIC Equipment Room Ambient Temperature - High	2	1	1, 2, 3	27
f. RCIC Equipment Room Δ Temp. - High	2	1	1, 2, 3	27
g. Main Steam Line Tunnel Ambient Temperature - High	2	1	1, 2, 3	27
h. Main Steam Line Tunnel Δ Temp. - High	2	1	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
6. RHR SYSTEM ISOLATION (continued)				
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	5	2	1, 2, 3	30
f. Drywell Pressure - High	10	2	1, 2, 3	30
g. Manual Initiation	10	1/group	1, 2, 3	26

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

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

ACTION 21 - Close the affected system isolation valve(s) within one hour or:

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

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

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

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

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function 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.

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

ACTION 28 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment operating within one hour. Initiate and maintain the Fuel Building Ventilation System in the Filtration mode of operation with (6) hours.

ACTION 29 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment operating within one hour. Initiate and maintain annulus mixing system with the reactor building annulus exhaust to at least one operating standby gas treatment train within (12) hours.

ACTION 30 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

NOTES

- * When handling irradiated fuel in the containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.
- ** May be bypassed with reactor steam pressure $\leq$ 1043 psig and all turbine stop valves less than 90% open and reactor MODE SWITCH not in RUN.
- *** The valve groups listed are designated in Tables 3.6.4-1 and 3.6.5.2-1.
- \# During CORE ALTERNATIONS and operations with a potential for draining the reactor vessel.

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.
(b) Also actuates the standby gas treatment system.
(c) Also actuates the main control room air conditioning system in the emergency mode of operation.
(d) Also trips and isolates the air removal pumps.
(e)
(f) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.5.2.1.
(g) Manual initiation of SLCS pump C001B closes G33-F001, and anual initiation of SLCS pump C001A closes G33-F004.
(h) Requires RCIC system steam supply pressure-low coincident with drywell pressure-high.
(i) Also starts the Fuel Building Exhaust Filter Trains A and B.
(j) Also starts the Annulus Mixing System.
(k) Also actuates the containment hydrogen analyzer/monitor recorder.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥-45.5 inches*	≥-47.7 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Containment Purge Isolation Radiation - High	≤ ( ) mR/hr**	≤ ( ) mR/hr**)
d. Manual Initiation	NA	NA
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-145.5 inches*	≥-147.7 inches
b. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3 x full power background
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 173** psid	≤ 178** psid
e. Condenser Vacuum - Low	≥ 8.5 inches Hg. vacuum	≥ 7.6 inches Hg. vacuum
f. Main Steam Line Tunnel Temperature - High	≤ ( )°F	≤ ( )°F
g. Main Steam Line Tunnel Δ Temp. - High	≤ ( )°F	≤ ( )°F
h. Main Steam Line Area Temperature - High	≤ ( )°F	≤ ( )°F
i. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ - 45.5 inches*	≥ - 47.5 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Fuel Building Ventilation Exhaust Radiation - High	≤ 4.5 mR/hr**	≤ 5.5 mR/hr**
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	≤ 35 mR/hr**	≤ 35 mR/hr**
e. Reactor Building Annulus Pressure Control System Air Flow-Low	≥ 600 cfm	≥ 485 cfm
f. Manual Initiation	NA	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 55 gpm	≤ 62.1 gpm
b. Δ Flow Timer	45 ± 2 seconds	≤ 47 seconds
c. Equipment Area Temperature - High	≤ ( )°F	≤ ( )°F
d. Equipment Area Δ Temp. - High	≤ ( )°F	≤ ( )°F
e. Reactor Vessel Water Level - Low Low, Level 2	≥ - 45.5 inches*	≥ - 47.7 inches
f. Main Steam Line Tunnel Ambient Temperature - High	≤ ( )°F	≤ ( )°F
g. Main Steam Line Tunnel Δ Temp. - High	≤ ( )°F	≤ ( )°F
h. SLCS Initiation	NA	NA
i. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	≤ 222" H_2O	≤ 230.5" H_2O
b. RCIC Steam Line Flow - High Timer	( ) ± ( ) seconds	≤ 10 seconds
c. RCIC Steam Supply Pressure - Low	≥ 60 psig	≥ 53 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
e. RCIC Equipment Room Ambient Temperature - High	≥ ( )°F ≤ ( )°F	≥ ( )°F ≤ ( )°F
f. RCIC Equipment Room Δ Temp. - High	≤ ( )°F	≤ ( )°F
g. Main Steam Line Tunnel Ambient Temperature - High	≤ ( )°F	≤ ( )°F
h. Main Steam Line Tunnel Δ Temp. - High	≤ ( )°F	≤ ( )°F
i. Main Steam Line Tunnel Temperature Timer	( ) ± ( ) seconds	( ) ± ( ) seconds
j. RHR Equipment Room Ambient Temperature - High	≤ ( )°F	≤ ( )°F
k. RHR Equipment Room Δ Temperature - High	≤ ( )°F	≤ ( )°F
l. RHR/RCIC Steam Line Flow - High	≤ 156" H_2O**	≤ 164.5" H_2O**
m. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
n. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. RHR SYSTEM ISOLATION		
a. RHR Equipment Area Ambient Temperature - High	$\leq$ ( )°F	$\leq$ ( )°F
b. RHR Equipment Area Δ Temperature - High	$\leq$ ( )°F	$\leq$ ( )°F
c. Reactor Vessel Water Level - Low, Level 3	$\geq$ 8.9 inches*	$\geq$ 8.3 inches
d. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ - 145.5 inches*	$\geq$ - 147.7 inches
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	$\leq$ 135 psig	$\leq$ 150 psig
f. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
g. Manual Initiation	NA	NA

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

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

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	< 10(a)
b. Drywell Pressure - High	≤ 10(a)
c. Containment Purge Isolation Radiation - High(b)	≤ 10(a))
d. Manual Initiation	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	1.0 */< 10(a)**
b. Main Steam Line Radiation - High(b)	1.0 */≤ 10(a)**
c. Main Steam Line Pressure - Low	1.0 */≤ 10(a)**
d. Main Steam Line Flow - High	0.5 */≤ 10(a)**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. Manual Initiation	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	≤ 10(a)
b. Drywell Pressure - High	≤ 10(a)
c. Fuel Building Ventilation Exhaust Radiation - High (b)	≤ 10(a)
d. Reactor Building Annulus Ventilation Sweep Exhaust Radiation - High (b)	≤ 10(a)
e. Reactor Building Annulus Pressure Control System Air Flow - Low	NA
f. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	≤ 10(a)(##)
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temp. - High	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≤ 10(a)
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

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

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

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

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

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

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

##Time delay of ( ) seconds.

###Time delay of ( ) seconds.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and #
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Containment Purge Isolation Radiation - High	S	M	R	1, 2, 3 and #
d. Manual Initiation	NA	M(a)	NA	1, 2, 3 and #
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	S	M	R	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
h. Main Steam Line Area Temperature-High (Turbine Building)	S	M	R	1, 2, 3
i. Manual Initiation	NA	M(a)	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 IN WHICH SURVEILLANCE REQUIRED
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and #
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Fuel Building Area Ventilation Exhaust Radiation - High High	S	M	R	1, 2, 3 and *
d. Reactor Building Annulus Ventilation Exhaust Radiation - High High	S	M	R	1, 2, 3 and *
e. Reactor Building Annulus Pressure Control System - Air Flow Low	S	M	A	1, 2, 3
f. Manual Initiation	NA	M(a)	NA	1, 2, 3 and *
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Δ Flow Timer	NA	M	Q	1, 2, 3
c. Equipment Area Temperature - High	S	M	R	1, 2, 3
d. Equipment Area Ventilation Δ Temp. - High	S	M	R	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 IN WHICH SURVEILLANCE REQUIRED
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (continued)				
e. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
h. SLCS Initiation	NA	M[b]	NA	1, 2, 3
i. Manual Initiation	NA	(M[a])(R)	NA	1, 2, 3
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	M	R	1, 2, 3
b. RCIC Steam Line Flow-High Timer	NA	M	Q	1, 2, 3
c. RCIC Steam Supply Pressure - Low	S	M	R	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R	1, 2, 3
e. RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
f. RCIC Equipment Room Δ Temp. - High	S	M	R	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 IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (continued)				
g. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
i. Main Steam Line Tunnel Temperature Timer	NA	M	Q	1, 2, 3
j. RHR Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
k. RHR Equipment Room Δ Temp. - High	S	M	R	1, 2, 3
l. RHR/RCIC Steam Line Flow-High	S	M	R	1, 2, 3
m. Drywell Pressure-High	S	M	R	1, 2, 3
n. Manual Initiation	NA	M(a)	NA	1, 2, 3
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	S	M	R	1, 2, 3
b. RHR Equipment Area Δ Temp. - High	S	M	R	1, 2, 3
c. RHR/RCIC Steam Line Flow - High	S	M	R	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 IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION (continued)				
d. Reactor Vessel Water Level - Low, Level 3	S	M	R	1, 2, 3
e. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
f. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	M	R	1, 2, 3
g. Drywell Pressure - High	S	M	R	1, 2, 3
h. Manual Initiation	NA	M(a)	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.

**When the Reactor Mode Switch is in RUN and any turbine stop valve is greater than 90% open.

#During CORE ALTERATION and operations with a potential for draining the reactor vessel.

(a) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system isolation.

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

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

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

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

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

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

   2. 72 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 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 FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. LPCS Pump Discharge Flow-Low (Bypass)	1	1, 2, 3, 4*, 5*	31
d. Reactor Vessel Pressure-Low (LPCS Permissive)	1	1, 2, 3	32
		4*, 5*	33
e. Reactor Vessel Pressure-Low (LPCI Permissive)	1	1, 2, 3	32
		4*, 5*	33
f. LPCI Pump A Start Time Delay Relay	(1)	1, 2, 3, 4*, 5*	32
g. LPCI Pump A Discharge Flow-Low (Bypass)	(1)	1, 2, 3, 4*, 5*	31
h. LPCS Pump Sart Time Delay Relay	(1)	1, 2, 3, 4*, 5*	32
i. Manual Initiation	1/system	1, 2, 3, 4*, 5*	35
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	32
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	32
g. Manual Initiation	2	1, 2, 3	35

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
B. DIVISION 2 TRIP SYSTEM			
1. RHR B & C (LPCI MODE)			
a. Reactor Vessel Water Level - Low, Low Low, Level 1	2[(b)]	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30
c. Reactor Vessel Pressure-Low (LPCI Permissive)	1/valve	1, 2, 3	32
		4*, 5*	33
d. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
e. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	31
f. LPCI Pump C Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
g. Manual Initiation	1	1, 2, 3, 4*, 5*	35
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2[(b)]	1, 2, 3	30
b. Drywell Pressure - High	2[(b)]	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCI Pump (B and C) Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	32
f. Manual Initiation	2	1, 2, 3	35

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4[(b)]	1, 2, 3, 4*, 5*	36[(c)]
b. Drywell Pressure - High	4[(b)]	1, 2, 3	36[(e)]
c. Reactor Vessel Water Level-High, Level (8)	2[(c)]	1, 2, 3, 4*, 5*	32
d. Condensate Storage Tank Level-Low	2[(d)]	1, 2, 3, 4*, 5*	37
e. Suppression Pool Water Level-High	2[(d)]	1, 2, 3, 4*, 5*	37
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	3
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1	1, 2, 3, 4*, 5*	35

TRIP FUNCTION	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM OPERABLE CHANNELS	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. LOSS OF POWER					
1. Division 1 and 2					
a. 4.16 kv Standby Bus Undervoltage (Sustained Undervoltage)	3/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	38
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	3/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	38
2. Division 3					
a. 4.16 kv Standby Bus Undervoltage	4/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	38

See footnotes on next page

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip function in the tripped condition provided at least one other OPERABLE channel in the same trip function is monitoring that parameter.
(b) Also actuates the associated division diesel generator (and the suppression pool makeup system).
(c) Provides signal to close HPCS injection valves only.
(d) Provides signal to open HPCS suppression pool suction valve only.
(e) This trip function logic is one out of two taken twice. Therefore, each one out of two logic is defined as a separate trip system for HPCS when complying with ACTION 36.
* When the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when ESF equipment is required to be OPERABLE.
# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to (100) psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

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

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

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour. Restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS valve or ECCS inoperable.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

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

ACTION 37 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable.

ACTION 38 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

*The provisions of Specification 3.0.4 are not appliable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-145.5 inches*	≥-147.7 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. LPCS Pump Discharge Flow-Low	≥ 875 gpm	≥ 750 gpm
d. Reactor Vessel Pressure-Low	≥ ( ) psig, decreasing	≥ ( ) psig, decreasing
e. Reactor Vessel Pressure-Low	≥ ( ) psig, decreasing	≥ ( ) psig, decreasing
f. LPCI Pump A Start Time Delay Relay	≤ 5 seconds	≤ 7 seconds
g. LPCI Pump A Discharge Flow-Low	≥ 900 gpm	≥ 750 gpm
h. LPCS Pump Start Time Delay Relay	≤ ( ) seconds	≤ 2 seconds
i. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-145.5 inches*	≥-147.7 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. ADS Timer	≥ 90, ≤ 105 seconds	≥ 90, ≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 8.9 inches*	≥ 8.3 inches
e. LPCS Pump Discharge Pressure-High	≥ 145 psig, increasing	≥ 125 psig, increasing
f. LPCI Pump A Discharge Pressure-High	≥ 125 psig, increasing	≥ 115 psig, increasing
g. Manual Initiation	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-145.5 inches*	≥-147.7 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Pressure-Low	≥ ( ) psig, decreasing	≥ ( ) psig, decreasing
d. LPCI Pump (B) Start Time Delay Relay	≤ 5 seconds	≤ 7 seconds
e. LPCI Pump Discharge Flow-Low	≥ ( ) gpm	≥ ( ) gpm
f. LPCI Pump C Start Time Delay Relay	≤ ( ) seconds	≤ 2 seconds
g. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥-145.5 inches*	≥-147.7 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. ADS Timer	≥ 90, ≤ 105 seconds	≥ 90, ≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 8.9 inches*	≥ 8.3 inches
e. LPCI Pump (B and C) Discharge Pressure-High	≥ 125 psig, increasing	≥ 115 psig, increasing
f. Manual Initiation	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - (Low Low, Level 2)	≥-45.5 inches*	≥-47.7 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Water Level - High, Level (8)	≤ 52 inches*	≤ 52.6 inches
d. Condensate Storage Tank Level - Low	≥ X+3 inches**	≥ X inches**
e. Suppression Pool Water Level - High	≤ Y-1.1 inches#	≤ Y inches#
f. Pump Discharge Pressure - High	≥ 140 psig increasing	≥ 125 psig increasing
g. HPCS System Flow Rate - Low	≥ 500 gpm	≥ 625 gpm
h. Manual Initiation	NA	NA
D. LOSS OF POWER		
1. Division 1 and 2		
a. 4.16 kv Standby Bus Undervoltage (Loss of Voltage) (##))	a. 4.16 kv Basis - (2940)±(161) volts	(2940)±(315) volts
	b. 120 v Basis - (84)±(4.6) volts	(84)±(9) volts
	c. ≤ (10) sec. time delay	≤ (10) sec. time delay
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	a. 4.16 kv Basis - (3727)±(9) volts	(3727)±(21) volts
	b. 120 v Basis - (106.5)±(0.25) volts	(106.5)±(0.60) volts
	c. (10)±(0.5) sec. time delay	(10)±(1) sec. time delay

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER (continued)		
2. Division 3		
a. 4.16 kv Standby Bus Undervoltage	a. 4.16 kv Basis - (3727)±(9) volts	(3727)±(21) volts
	b. 120 v Basis - (106.5)±(0.25) volts	(106.5)±(0.60) volts
	c. (10)±(0.5) sec. time delay	(10)±(1) sec. time delay

*See Bases Figure B 3/4 3-1.
**Xis the minimum CST level to allow time to switch to pool suction.
#Yis pool high water level defined as 5 inches above normal water level.
##These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	$\leq$ 37
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
a. Pumps A and B	$\leq$ 37
b. Pump C	$\leq$ 37
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	$\leq$ 27
5. LOSS OF POWER	NA

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Pressure-Low	S	M	R	1, 2, 3, 4*, 5*
d. LPCI Pump (B) Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	S	M	R	1, 2, 3, 4*, 5*
f. LPCI Pump C Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
g. Manual Initiation	NA	M(b)	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	S	M	R(a)	1, 2, 3
f. Manual Initiation	NA	M(b)	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - (Low Low, Level 2)	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Water Level-High, Level (8)	S	M	R(a)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R(a)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(a)	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	S	M	R	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	S	M	R	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	M(b)	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Division 1 and 2				
a. 4.16 kv Standby Bus Undervoltage (sustained undervoltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kv Standby Bus Undervoltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. Division 3				
a. 4.16 kv Standby Bus Undervoltage	NA	NA	R	1, 2, 3, 4**, 5**

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

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

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

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

((b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.)

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

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

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, declare the trip system(s) inoperable.

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

    1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within one hour.

    2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

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

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

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	$\geq$-45.5 inches*	$\geq$-47.7 inches
2. Reactor Vessel Pressure - High	$\leq$ 1127 psig	$\leq$ 1142 psig

*See Bases Figure B3/4 3-1.

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

*Calibrate trip unit at least once per 31 days.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (40)% of RATED THERMAL POWER.

ACTION:

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

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

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

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

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

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or reduce THERMAL POWER to less than 40% of RATED THERMAL POWER within the next 6 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The time allotted for breaker arc suppression, 100 ms, shall be verified at least once per 60 months.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

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

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

(b)This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 250* psig, equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

*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.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

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

TABLE 3.3.4.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

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

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	M*	R#
2. Turbine Control Valve-Fast Closure	M*	R#
3. Turbine First Stage Pressure	M*	R#

*Including trip system logic testing.

#Calibrate trip unit at least once per 31 days.

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 Water Level - (Low Low, Level 2)	2	50
b. Reactor Vessel Water Level - High, Level (8)	2[(b)]	51
c. Condensate Storage Tank Water Level - Low	2[(c)]	52
d. Suppression Pool Water Level - High	2[(c)]	52
e. Manual Initiation	1[(d)]	53

(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
ACTUATION INSTRUMENTATION

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

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

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

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip System requiremnt, declare the RCIC system inoperable.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, place at least one inoperable channel in the tripped condition within one hour or delcare the RCIC system inoperable.

ACTION 53 - 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 RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level - (Low Low, Level 2)	$\geq$ -45.5 inches*	$\geq$ -47.7 inches
b. Reactor Vessel Water Level - High, Level (8)	$\leq$ 52 inches*	$\leq$ 54.2 inches
c. Condensate Storage Tank Level - Low	$\geq$ 14 inches	$\geq$ 9 inches
d. Suppression Pool Water Level - High	$\leq$ 5 inches	$\leq$ 21 inches
e. 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 Water Level - (Low Low, Level 2)	S	M	R[(a)]
b. Reactor Vessel Water Level - High, Level (8)	S	M	R
c. Condensate Storage Tank Level - Low	S	M	R
d. Suppression Pool Water Level - High	S	M	R
e. Manual Initiation	NA	M[(b)]	NA

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

((b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.)

INSTRUMENTATION

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 PATTERN CONTROL SYSTEM			
a. Low Power Setpoint	2	1, 2	60
b. High Power Setpoint	2	1, 2	60
2. APRM			
a. Flow Biased Neutron Flux - Upscale	6	1	61
b. Inoperative	6	1, 2, 5	61
c. Downscale	6	1	61
d. Neutron Flux - Upscale, Startup	6	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(a)	3	2	61
	2	5	61
b. Upscale(b)	3	2	61
c. Inoperative(b)	3	2	61
	2	5	61
d. Downscale(c)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in(d)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(d)	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5*	62
b. Scram Trip Bypass	2	1, 2, 5*	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

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

ACTION 61 - With the number of OPERABLE Channels:

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

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

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

NOTES

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

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

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

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

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

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. **ROD PATTERN CONTROL SYSTEM**		
a. Low Power Setpoint	27.5 ± 3 of RATED THERMAL POWER	27.5 ± 7.5 of RATED THERMAL POWER
b. High Power Setpoint	62.5 ± 3 of RATED THERMAL POWER	62.5 ± 7.5 of RATED THERMAL POWER
2. **APRM**		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. **SOURCE RANGE MONITORS**		
a. Detector not full in	NA	NA
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 2 cps
4. **INTERMEDIATE RANGE MONITORS**		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 division of full scale	≤ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 division of full scale	≥ 3/125 division of full scale
5. **SCRAM DISCHARGE VOLUME**		
a. Water Level-High	≤ 32.5 inches	≤ 34 inches
b. Scram Trip Bypass	NA	NA
6. **REACTOR COOLANT SYSTEM RECIRCULATION FLOW**		
a. Upscale	≤ 108% of rated flow	≤ (111)% of rated flow

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

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

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

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

e. Includes reactor manual control multiplexing system input.

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

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

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	2/intake	1,2,3,5 and *	$\leq$ 900 cpm	70
2. Area Monitors				
a. Criticality Monitors Spent Fuel Storage Pool	1	#	$\leq$ 15 mR/hr[(a)]	71
b. Control Room Direct Radiation Monitor	1	At all times	$\leq$ ( ) mR/hr[(a)]	71
3. Main Condenser Offgas Treatment System Effluent Monitoring System				
a. Noble Gas Activity Monitor - (Providing Alarm and Automatic Termination of Release)	1	**	(Later)	72
4. Condenser Air Ejector Radioactivity Monitor (Prior to Input to Holdup System)				
a. Noble Gas Activity Monitor	1	**	(Later)	73
b. Flow Rate Monitor	1	*	(Later)	74

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATION

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

**When the off-gas treatment system is operating.

(a) Alarm only.

#With fuel in the spent fuel storage pool.

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 70 -

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

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

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

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

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

a. The offgas system is not bypassed, and

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

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

ACTION 74 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Main Control Room Ventilation Radiation Monitor	S	M	R	1, 2, 3, 5 and *
2. Area Monitors -				
a. Criticality Monitors Spent Fuel Storage Pool	S	M	R	#
b. Control Room Direct Radiation Monitor	S	M	R	At all times
3. Main Condenser Offgas Treatment System Effluent Monitoring System				
a. Noble Gas Activity Monitor - (Providing Alarm and Automatic Termination of Release)	S	M	R	At all times
4. Condenser Air Ejector Radioactivity Monitor				
a. Noble Gas Activity Monitor	S	M	R	1, 2, 3, 4, 5**
b. Flow Rate Monitor	S	M	R	1, 2, 3, 4, 5**

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

*When irradiated fuel is being handled in the secondary containment.
**When the off-gas treatment system is operating.
#With fuel in the spent fuel storage pool.

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.

TABLE 3.2.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs		
a. Reactor Bldg MAT. EL 70'0"	0 ± 1.0 g	1
b. Reactor Bldg Ext Shield Wall EL 232'0"	0 ± 1.0 g	1
c. Reactor Bldg Drywell EL 151'0"	0 ± 1.0 g	1
d. Free Field - Grade Level	0 ± 1.0 g	1
2. Triaxial Peak Accelerographs		
a. Reactor Bldg SLCS Storage Tank	0 ± 10.0 g	1
b. Reactor Bldg - RHR Inj. Piping	0 ± 10.0 g	1
c. Aux. Bldg Service Water Piping	0 ± 10.0 g	1
3. Triaxial Seismic Switches		
a. Reactor Bldg MAT. El 70'0"	0.025 to 0.25 g (adjustable)	1[a]
4. Triaxial Response-Spectrum Recorders		
a. Reactor Bldg MAT. EL 70'0"	0 ± 2 g	1[a]
b. Reactor Bldg (Floor EL 141'0")	0 ± 2 g	1
c. Auxiliary Bldg MAT. EL 70'0"	0 ± 2 g	1
d. Auxiliary Bldg Floor EL 141'0"	0 ± 2 g	1

[a] With reactor control room indication and annuciation.

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs			
a. Reactor Bldg. MAT EL 70'0"	M	SA	R
b. Reactor Bldg. Exit Shield Wall EL 232'0"	M	SA	R
c. Reactor Bldg. Drywell EL 151'0"	M	SA	R
d. Free Field-Grade Level	M	SA	R
2. Triaxial Peak Accelerographs			
a. Reactor Bldg. SLCS Storage Tank	NA	NA	R
b. Reactor Bldg. - RHR Inj. Piping	NA	NA	R
c. Aux. Bldg. Service Water Piping	NA	NA	R
3. Triaxial Seismic Switches			
a. Reactor Bldg. MAT EL 70'0"	M[(a)]	SA	R
4. Triaxial Response-Spectrum Recorders			
a. Reactor Bldg. MAT EL 70'0"	M	SA	R
b. Reactor Bldg. (Floor EL 141'0"	NA	NA	R
c. Auxiliary Bldg. MAT EL 70'0"	NA	NA	R
d. Auxiliary Bldg. Floor EL 141'0"	NA	NA	R

[(a)] Except seismic trigger.

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. 33 ft.	1
2. Elev. 150 ft.	1
b. Wind Direction	
1. Elev. 33 ft.	1
2. Elev. 150 ft.	1
c. Air Temperature Difference	
1. Elev. 33/150 ft.	1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 33 ft.	D	SA
2. Elev. 150 ft.	D	SA
b. Wind Direction		
1. Elev. 33 ft.	D	SA
2. Elev. 150 ft.	D	SA
c. Air Temperature Difference		
1. Elev. 33/150 ft.	D	SA

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With 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		READOUT LOCATION*	MINIMUM INSTRUMENTS OPERABLE PER PANEL
1. Reactor Vessel Pressure		RSP1, RSP2	1
2. Reactor Vessel Water Level		RSF1, RSP2	1
3. Safety/Relief Valve Demand Position, (3) valves		RSP1, RSP2	1/valve
4. Suppression Pool Water Level		RSP1, RSP2	1
5. Suppression Pool Water Temperature		RSP1, RSP2	1
6. Drywell Pressure		RSP1, PRS2	1
7. Drywell Temperature		RSP1, RSP2	1
8. RHR System Flow:	Loop A	RSP1	1
	Loop B	RSP2	1
	Loop C	RSP2	1
9. RHR Hx Cooling Water System Flow:	Loop A	RSP1	1
	Loop B	RSP2	1
11. RCIC System Flow		RSP1	1
12. RCIC Turbine Speed		RSP1	1

*RSPI - Remote Shutdown Panel Division I
RSPI - Remote Shutdown Panel Division II

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure		M	R
2. Reactor Vessel Water Level		M	R
3. Safety/Relief Valve Position		M	NA
4. Suppression Pool Water Level		M	R
5. Suppression Pool Water Temperature		M	R
6. Drywell Pressure		M	R
7. Drywell Temperature		M	R
8. RHR System Flow:	Loop A	M	R
	Loop B	M	R
	Loop C	M	R
9. RHR Hx Cooling Water System Flow:	Loop A	M	R
	Loop B	M	R
11. RCIC System Flow		M	R
12. RCIC Turbine Speed		M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERABLE CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2	80
2. Reactor Vessel Water Level	2	1	1,2	80
a. Wide Range	2	1	1,2	80
b. Fuel Zone	2	1	1,2	80
3. Suppression Pool Water Level	2	1	1,2	80
4. Suppression Pool Water Temperature	2/sector	1/sector	1,2	80
5. Primary Containment Air Temperature	2	1	1,2	80
6. Primary Containment Pressure	2	1	1, 2, 3	81
7. Drywell Pressure	2	1	1,2	80
8. Drywell Air Temperature	2	1	1,2	80
9. Drywell and Primary Containment Hydrogen Concentration Analyzer and Monitor	2	1	1,2	80
10. Safety/Relief Valve Position Indicators	2/valve	1/valve	1,2	80
11. In-Core Thermocouples	4/1 per core quadrants)	2/1 each of 2 core quadrants)	1,2 1,2	80
12. Area Radiation#				
a. Primary Containment/Drywell Area	2	1	1, 2, 3	81
b. Secondary Containment Fuel Building	2	1	1, 2, 3	81
c. Secondary Containment Auxiliary Building	2	1	1, 2, 3	81
13. Containment Ventilation Exhaust Monitor#	1	1	1,2,3	81
14. Auxiliary Building/Fuel Handling Area Ventilation Monitor#	1	1	1,2,3	81
15. Standby Gas Treatment System Exhaust Monitor#	1	1	1,2,3	81
16. Off-Gas and Radwaste Building Ventilation Exhaust Monitor#	1	1	1,2,3	81
17. Turbine Building Ventilation/Hogging Pump Exhaust Monitor#	1	1	1,2,3	81

#High range noble gas monitors.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATIONS

ACTION STATEMENTS

Action 80 -

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

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

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

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

b. Prepare and submit 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	M	R	1, 2
a. Wide Range	M	R	1, 2
b. Fuel Zone	M	R	1, 2
3. Suppression Pool Water Level	M	R	1, 2
4. Suppression Pool Water Temperature	M	R	1, 2
5. Primary Containment Air Temperature	M	R	1, 2
6. Primary Containment Pressure	M	R	1, 2
7. Drywell Pressure	M	R	1, 2
8. Drywell Air Temperature	M	R	1, 2
9. Drywell and Primary Containment Hydrogen Concentration Analyzer and Monitor	M	Q*	1, 2
10. Safety/Relief Valve Position Indicators	M	R	1, 2
11. In-Core Thermocouples	M	R	1, 2
12. Area Radiation#			
a. Primary Containment/Drywell Area	M	R	1, 2
b. Secondary Containment Fuel Building	M	R	1, 2
c. Secondary Containment Auxiliary Building	M	R	1, 2
13. Containment Ventilation Exhaust Monitor#	M	R	1, 2, 3
14. Auxiliary Building/Fuel Handling Area Ventilation Monitor#	M	R	1, 2, 3
15. Standby Gas Treatment System Exhaust Monitor#	M	R	1, 2, 3
16. Off-Gas and Radwaste Building Ventilation Exhaust Monitor#	M	R	1, 2, 3
17. Turbine Building Ventilation/Hogging Pump Exhaust Monitor	M	R	1, 2, 3

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

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

#High range noble gas monitors.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

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

a. In OPERATIONAL CONDITION 2*, three,

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. Performance of a:

   1. CHANNEL CHECK at least once per:

      a) 12 hours in CONDITION 2*, and

      b) 24 hours in CONDITION 3 or 4.

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

b. Performance of a CHANNEL FUNCTIONAL TEST:

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

   2. At least once per 31 days.

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

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

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

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

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

a. Recalibration of the LPRM detectors, and

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.7.7 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 when required for the above applicable monitoring or calibration functions.

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

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the number of OPERABLE Function A or Function B fire detection instruments less than the Minimum Instruments OPERABLE requirement of Table 3.3.7.8-1.

a. Within 1 hour, establish a fire watch patrol to inspect the zone(s) with the Function A or room(s) with Function B inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, steam tunnel or drywell, then inspect the primary containment at least once per 8 hours or monitor the containment, steam tunnel and/or drywell air temperature at least once per hour at the locations listed in Specification 3.7.8, 4.6.1.8 and 4.6.2.6.

b. Restore the minimum number of instruments to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.7.8-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		MINIMUM INSTRUMENTS OPERABLE*		
		HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
I. CONTROL BUILDING ZONE				
SD-1	HVAC ROOM, EL 115'0" & 116'0"			6/0
SD-2	HPCS SWGR, EL 115'0" & 116'0"			3/0
SD-3	BATTERY ROOMS (3) & DC EQUIP RMS, EL 115'0" & 116'0"			8/0
SD-4	HVAC ROOM, EL 98'0"			6/0
SD-5	STBY SWGR ROOM A, EL 98'0"			3/0
SD-6	STBY SWGR ROOM B, EL 98'0"			3/0
SD-15	HVAC ROOM 1A, EL 70'0"			2/0
SD-16	HVAC ROOM 1B, EL 70'0"			2/0
SD-17	CABLE VAULT, EL 70'0"			0/2
SD-18	CABLE VAULT, EL 70'0"			0/3
SD-19	CABLE VAULT, EL 70'0"			0/7
SD-20	CABLE CHASES, EL 70'0"			14/0
SD-50	CABLE CHASES, EL 98'0"			9/0
SD-54	CABLE CHASES, EL 116'0"			10/0
SD-60	125 VDC SWGR & BATT CHGR EL 115'0" & 116'0"			10/0
SD-61	GENERAL AREA, EL 98'0"			8/0
SD-125 thru SD-151, SD-158	PGCC FLOOR PANELS, EL 136'0"	(---------LATER---------)		
SD-152	NON PANEL MODULE AREA NORTH, EL 135'0"			10/0
SD-153	NON PANEL AREA SOUTH, EL 135'0'"			10/0
SD-154	GENERAL AREA, EL 136'0'"			84/0
SD-162	REMOTE SHUTDOWN PANEL DIV 1, EL 98'0"	(--------LATER----------)		
SD-163	REMOTE SHUTDOWN PANEL DIV 2, EL 98'0"	(--------LATER----------)		
FD-26	CHARCOAL FILTER 1HVC*FLT3B, EL 115'0"	0/1		
FD-27	CHARCOAL FILTER 1HVC*FLT3A, EL 115'0"	0/1		

* (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.

TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		MINIMUM INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
II.	REACTOR BUILDING ZONE			
SD-57	#CONTAINMENT AREA, EL 114'0"			8/0
SD-102	ANNULUS AREA, EL 186'3"			27/0
SD-104	#CONTAINMENT AREA, EL 186'3"			17/0
SD-117	#CONTAINMENT AREA, EL 162'3"			6/0
SD-119	#CONTAINMENT AREA, EL 141'0"			13/0
SD-156	#CONTAINMENT AREA, EL 95'9"			2/0
FD-13	#RECIRC PUMPS - DRYWELL, EL 70'0" & 98'0"	2/0		
III.	AUXILIARY BUILDING ZONE			
SD-28	HPCS PUMP ROOM EL 70'0"			1/0
SD-29	RHR PUMP ROOM B, EL 70'0"			2/0
SD-30	RHR PUMP ROOM C, EL 70'0"			2/0
SD-31	RHR PUMP ROOM D, EL 70'0"			2/0
SD-32	LPCS PUMP ROOM, EL 70'0"			1/0
SD-43	GENERAL AREA WEST, EL 95'0"			2/0
SD-49	GENERAL AREA, EL 141'0"			11/0
SD-52	GENERAL AREA EAST, EL 114'0"			5/0
SD-53	GENERAL AREA WEST, EL 114'0"			4/0
SD-55	PASS ROOM EL 114'0"			1/0
SD-96	RCIC PUMP ROOM EL 70'0"			0/2
SD-97	GENERAL AREA, EL 70'0"			5/0
SD-98	GENERAL AREA EAST, EL 95'9"			2/0
SD-99	GENERAL AREA WEST, EL 95'9"			2/0
SD-100	GENERAL AREA WEST, EL 95'9"			2/0
SD-101	STANDBY GAS TREATMENT ROOM "B", EL 141'0"			4/0

* (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.

TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		MINIMUM INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
III. AUXILIARY BUILDING ZONE (Continued)				
SD-103	STANDBY GAS TREATMENT ROOM "A", EL 141'0"			4/0
SD-106	ANNULUS MIXING FAN AREA, EL 171'0"			3/0
FD-28	CONTAINMENT PURGE FILTER AREA, EL 171'0"	0/1		
FD-33	STANDBY GAS TREATMENT FILTER "B", EL 141'0"	0/1		
FD-34	STANDBY GAS TREATMENT FILTER "A", EL 141'0"	0/1		
IV. FUEL BUILDING ZONE				
SD-33	FUEL POOL COOLING PUMP AREAS, EL 70'0"			2/0
SD-44	1ENS*SWG 3A & 4A AREA, EL 95'0"			6/0
SD-59	GENERAL AREA, EL 113'0"			13/0
SD-91	GENERAL AREA, EL 70'0"			7/0
SD-94	NEW FUEL RECEIVING AREA, EL 95'0"			2/0
SD-110	FUEL POOL PURIFICATION & BACKWASH PUMP AREAS, EL, 70'0"			3/0
SD-111	FUEL POOL COOLER (A & B) AREAS, EL 95'0"			2/0
SD-121	CHARCOAL FILTER "A" ROOM, EL 148'0"			2/0
SD-123	CHARCOAL FILTER "B" ROOM, EL 148'0"			2/0
SD-124	1RMS*CAB101 AREA, EL 148'0"			4/0
SD-155	GENERAL AREA, EL 113'0"			4/0
FD-35	CHARCOAL FILTER "A" ROOM, EL 148'0"	0/1		
FD-36	CHARCOAL FILTER "B" ROOM, EL 148'0"	0/1		

* (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.

TABLE 3.3.7.8-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION		MINIMUM INSTRUMENTS OPERABLE* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
V.	ELECTRICAL TUNNELS ZONE			
SD-79	GENERAL AREA, EL 67'6"			3/3
SD-80	GENERAL AREA, EL 67'6"			3/3
SD-81	GENERAL AREA, EL 67'6"			6/5
SD-82	GENERAL AREA, EL 67'6"			6/6
SD-83	GENERAL AREA, EL 70'0"			0/10
VI.	PIPE TUNNEL ZONE			
SD-86	GENERAL AREA, EL 70'0"			0/9
SD-87	GENERAL AREA, EL 67'6"			0/4
SD-88	GENERAL AREA, EL 67'6"			0/5
SD-89	GENERAL AREA, EL 67'6"			0/8
VII.	DIESEL GENERATOR BUILDING ZONE			
SD-105	GENERAL AREA, EL 98'0"			3/0
FD-16	DIESEL ROOM DIV. II, EL 98'0"	0/4		
FD-17	DIESEL ROOM DIV III, EL 98'0"	0/4		
FD-18	DIESEL ROOM DIV I, EL 98'0	0/4		
VIII.	STANDBY SERVICE WATER PUMP HOUSE ZONE			
SD-72	STANDBY SERVICE WATER DIV 1 PUMP AREA			2/0
SD-73	STANDBY SERVICE WATER DIV 2 PUMP AREA			2/0

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

3.3.7.9 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

a. CHANNEL CHECK at least once per 24 hours,

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

c. CHANNEL CALIBRATION at least once per 18 months.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

c. The provisions of Specifications 3.0.3, 3.0.4, and (6.9.1.9.b) are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

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

TABLE 3.3.7.10-1 (Continued)

TABLE NOTATION

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

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1 and (4.11.1.1.2, and)

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

Otherwise, suspend release of radioactive effluents via this pathway.

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

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

TABLE 4.3.7.10-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3.7.10-1 (Continued)

TABLE NOTATION

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

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

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annuciation 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 CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used. (Operating plants may substitute previously established calibration procedures for this requirement.)

(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 which continuous, periodic, or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.11-1.

ACTION:

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

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

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9.b are not applicable.

SUREVEILLANCE REQUIREMENTS

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

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*(The alarm/trip setpoints for the Explosive Gas Mixture in the Main Condenser Offgas Treatment System, and the Main Condenser Offgas Treatment Noble Gas Activity Monitor are set in accordance with Specification 3.11.2.6 and 3.11.2.7, respectively.)

TABLE 3.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1.	Main Condenser Offgas Treatment System Effluent Monitoring System			
	a. Noble Gas Activity Monitor - (Providing Alarm and Automatic Termination of Release)	1	*	123
2.	Main Condenser Offgas Treatment System Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)			
	a. Hydrogen Monitor (downstream of the recombiner)	1	**	125
3.	Main Plant Exhaust Duct Monitoring System			
	a. Noble Gas Activity Monitor	1	*	123
	b. Iodine Sampler	1	*	127
	c. Particulate Sampler	1	*	127
	d. Effluent System Flow Rate Monitor	1	*	122
	e. Sampler Flow Rate Monitor	1	*	122

TABLE 3.3.7.11-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Fuel Building Exhaust Duct Monitoring System			
a. Noble Gas Activity Monitor	1	*	123
b. Iodine Sampler	1	*	127
c. Particulate Sampler	1	*	127
d. Flow Rate Monitor	1	*	122
e. Sampler Flow Rate Monitor	1	*	122
5. Radwaste Building Ventilation Exhaust Duct Monitoring System			
a. Noble Gas Activity Monitor	1	*	123
b. Iodine Sampler	1	*	127
c. Particulate Sampler	1	*	127
d. Flow Rate Monitor	1	*	122
e. Sampler Flow Rate Monitor	1	*	122
6. Condenser Air Ejector Radioactivity Monitor (Prior to Input to Holdup System)			
a. Noble Gas Activity Monitor	1	***	121

TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

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

ACTION 121 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 72 hours provided:

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.

ACTION 122 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 123 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 125 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 127 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3.7.11-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EFFLUENT MONITORING SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	R(3)	Q(1)	*
2. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion					
a. Hydrogen Monitor (downstream of the recombiner)	D	N.A.	Q(4)	M	**
3. MAIN STACK MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3.7.11-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. FUEL BUILDING EXHAUST DUCT MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
5. RADWASTE BUILDING VENTILATION EXHAUST DUCT MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
6. CONDENSER AIR EJECTOR RADIOACTIVITY MONITOR					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***

TABLE 4.3.7.11-1 (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 automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

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

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

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

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

INSTRUMENTATION

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

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

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

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

a.- At least once per 7 days by:

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

a) For the overspeed protection control system;

1) Four high pressure turbine control valves, and

2) Four low pressure turbine interceptor valves

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

1) Four high pressure turbine throttle stop valves,

2) Four high pressure turbine reheat stop valves,

3) Four high pressure turbine control valves, and

4) Four low pressure turbine interceptor valves.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

INSTRUMENTATION

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The plant systems actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.13-2.

APPLICABILITY: As shown in Table 3.3.13-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.13-2, declare the channel inoperable and take the ACTION required by Table 3.3.13-1.

b. With one or more plan systems actuation instrument channels inoperable, take the ACTION required by Table 3.3.13-1.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each plant system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.9.1-1.

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

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

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

TABLE 3.3.9-1 (Continued)

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

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

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

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

ACTION 152 - Declare the Associated Containment Ventilation System inoperable.

ACTION 153 - 1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

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

#Provided this does not actuate the system.

TABLE 3.3.9-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B		
a. Drywell Pressure-High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
b. Containment-to-Annulus ΔP-High	$\leq$ 11.98" H_2O	$\leq$ 12.20" H_2O
c. Reactor Vessel Water Level-Low Low Low, Level 1	$\geq$-145.5 inches*	$\geq$-147.7 inches
d. Timer	50 ± 4 seconds	$\leq$ 60 seconds
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM		
a. Reactor Vessel Water Level-High, Level 8	$\leq$ 52 inches*	$\leq$ 53.5 inches

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

TABLE 4.3.9.1-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT VENTILATION SYSTEM - UNIT COOLER A AND B				
a. Drywell Pressure-High	S	M	Q	1, 2, 3
b. Containment-to-Annulus ΔP-High	S	M	Q	1, 2, 3
c. Reactor Vessel Water Level-Low Low Low, Level 1	S	M	Q	1, 2, 3
d. Timer	NA	M	R	1, 2, 3
2. FEEDWATER SYSTEM/MAIN TURBINE TRIP SYSTEM				
a. Reactor Vessel Water Level-High, Level 8	NA	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 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

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

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

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

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

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

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 loops are operating at the same flow control valve position.

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics.

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

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.4.

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F,* and:

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

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

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

Number of Valves	Function	Setpoint* (psig) ± 1%
7	Safety	1150
5	Safety	1165
4	Safety	1175
1	Relief	1103
8	Relief	1113
7	Relief	1123

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the open valve(s) within 2 minutes or if suppression pool average water temperature is (105)°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 1.8 ± 0.04 volts by performance of a:

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

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

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

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

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

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The relief valve function and 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) ± 1%		Relief Function Setpoint* (psig) ± 1%	
	Open	Close	Open	Close
1B21*RVF051D	1033	926	1103	1003
1B21*RVF051C	1073	936	1113	1013
1B21*RVF051B	1113	946	1113	1013
1B21*RVF051G	1113	946	1113	1013
1B21*RVF047F	1113	946	1113	1013

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. The drywell atmosphere particulate radioactivity monitoring system,

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

c. Either the drywell air coolers condensate flow rate monitoring system) or the drywell atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Drywell atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Drywell sump and pedestal floor sump drain flow monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

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

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

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

d. 1 gpm leakage at a reactor coolant system pressure of 1025 ± 15 (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 two other closed manual, deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

b. Monitoring the drywell and pedestal floor drain sump flow rates at least once per 12 hours,

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

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

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

a. At least once per 18 months, and

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

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

4.4.3.2.3 The 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.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NUMBER	FUNCTION
a) LPCS	1E21*AOVF006	LPCS Injection
	1E21*MOVF005	LPCS Injection
b) HPCS	1E22*AOVF005	HPCS Injection
	1E22*MOVF004	HPCS Injection
c) RCIC	1E51*AOVF065	RCIC Head Spray
	1E51*MOVF013	RCIC Head Spray
d) RHR	1E12*MOVF023	RHR Head Spray
	1E12*AOVF041A	LPCI A Injection
	1E12*MOVF042A	LPCI A Injection
	1E12*AOVF041B	LPCI B Injection
	1E12*MOVF042B	LPCI B Injection
	1E12*AOVF041C	LPCI C Injection
	1E12*MOVF042C	LPCI C Injection
	1E12*MOVF009	Shutdown Cooling A & B Suction
	1E12*MOVF008	Shutdown Cooling A & B Suction
	1RHS*V240	Shutdown Cooling A & B Suction

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

INSTRUMENT NUMBER	FUNCTION	ALARM SETPOINT
1E21*PTN054	LPCS Pump Discharge Pressure High	<582 psig
1E22*PTN052	HPCS Pump Suction Pressure High	≤82 psig
1E51*PTN052	RCIC Pump Suction Pressure High	≤83 psig
1E12*PTN053A	RHR A Pump Discharge Pressure High	≤482 psig
1E12*PTN053B	RHR B Pump Discharge Pressure High	≤482 psig
1E12*PTN053C	RHR C Pump Discharge Pressure High	≤482 psig
1E12*PTN057	RHR Pump Shutdown Cooling Suction Pressure High	≤195 psig

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 CONDITION 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

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. 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.

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 coolant, or, when the continuous recording conductivity monitor is inoperable for up to 31 days, obtaining an in-line conductivity measurement at least once per:

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

2. 24 hours at all other times.

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

1. 7 days, and

2. 24 hours whenever conductivity is greater than the limit in Table 3.4.4-1.

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

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

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

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

2. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or for more than 800 hours cumulative operating time in a consecutive 12-month period, or greater than 4.0 microcuries per gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

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

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses

ACTION (Continued)

and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

c. In OPERATIONAL CONDITION 1 or 2, with:

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

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

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

perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. 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.

*Not applicable during the startup test program.

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.

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.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

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

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; 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 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

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

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

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

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

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

Reactor Pressure in RPV Top Head (psig)

Curve(s) ( ) applicable for
service periods up to ( ) EFPY
(and contain margins of (10)°F
and (60) psig for possible
instrument errors).

(RPV Metal) Temperature (°F)

MINIMUM (REACTOR PRESSURE VESSEL METAL) TEMPERATURE REACTOR VESSEL PRESSURE
Figure 3.4.6.1-1

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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.

**The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITIONS 2 or 3 provided the valves are initially demonstrated OPERABLE by a cold functional test and the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.

REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

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

ACTION:

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

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

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.

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

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

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

3. At least 7 OPERABLE ADS valves.

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, each with a flow path capable of taking suction from the suppression pool and transferring the water to the reactor vessel.

2. At least 7 OPERABLE ADS valves.

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

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

*The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

#See Special Test Exception 3.10.5.

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

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

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

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

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

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

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

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

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

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

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

1) With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

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

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

ACTION: (Continued)

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

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

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

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

e. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:

1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$ 100 psig within the next 24 hours.

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

f. 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.

*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.

SURVEILLANCE REQUIREMENTS

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

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

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

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

1. LPCS pump develops a flow of at least 5010 gpm with 2 pump differential pressure greater than or equal to 301 psid.

2. LPCI pump develops a flow of at least 5050 gpm with a pump differential pressure greater than or equal to 119 psid.

3. HPCS pump develops a flow of at least 5010 gpm with a pump differential pressure greater than or equal to 386 psid.

c. For the LPCS, LPCI and HPCS systems, at least once per 18 months, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

d. For the HPCS system, at least once per 18 months, verifying that the suction is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal and on a suppression pool high water level signal.

*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.

SURVEILLANCE REQUIREMENTS (Continued)

e. For the ADS by:

1. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas 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 100 psig* and observing the expected change in the indicated valve position 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 accumulator backup compressed gas system low pressure alarm system and verifying an alarm setpoint of (   ) + (   ), -(   ) psig on decreasing pressure.

*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 shall be OPERABLE:

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

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

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

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

e. The high pressure core spray (HPCS) system with a flow path capable of taking suction from one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

1. From the suppression pool, or

2. When the suppression pool level is less than the limit or is drained, from the condensate storage tank containing at least 125,000 available gallons of water, equivalent to a level of 33.2%.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5*.

ACTION:

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

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

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 HPCS system shall be determined 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.e.

3/4.5.3 SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

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

b. In OPERATIONAL CONDITION 4 and 5* with a contained water volume of at least 115,879 ft^3, equivalent to a level of 16'2", except that the suppression pool level may be less than the limit or may be drained provided that:

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

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

3. The condensate storage tank contains at least (125,000) available gallons of water, equivalent to a level of 33.2%, and

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

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

ACTION: -

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

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

*The suppression pool 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 upper containment fuel pool gates are removed, the spent fuel pool gates are removed (when the cavity is flooded), and the water level is maintained within the limits of Specification 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

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

a. 19'6" at least once per 24 hours.

b. 16'2" at least once per 12 hours.

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

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

*The suppression pool 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.

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 containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 6.31 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 0.60 La.

b. At least once per 31 days by verifying that all 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, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

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

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

*See Special Test Exception 3.10.1

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

PRIMARY CONTAINMENT INTEGRITY - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.6.1.2 PRIMARY CONTAINMENT INTEGRITY - SHUTDOWN shall be maintained.

APPLICABILITY: OPERATIONAL CONDITION 5*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.2 PRIMARY CONTAINMENT INTEGRITY - SHUTDOWN shall be demonstrated:

a. Within 24 hours prior to and at least once per 31 days during OPERATIONAL CONDITION 5* by verifying that all containment penetrations shown in Table 3.6.4-1 which are not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during fuel handling accident conditions are closed by hatches, valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

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

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

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

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.3 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.26 percent by weight of the containment air per 24 hours at P_a, 6.31 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and all valves listed in Table 3.6.4-1, except for main steam line isolation valves* and valves which are hydrostatically leak tested per Table 3.6.4-1, subject to Type B and C tests when pressurized in accordance with Table 3.6.4-1 of Specification 3.6.4.

c.(*)Less than or equal to 11.5 scf per hour for any one main steam line through the isolation valve when tested at 6.31 psig.

d. A combined leakage rate of less than or equal to 30,000 cc/hr for all penetrations shown in Table 3.6.4-1 of Specification 3.6.4 as secondary containment bypass leakage paths when pressurized in accordance with Table 3.6.4-1

e. A combined leakage rate of less than or equal to 3 gpm for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a 6.94 psig.

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

ACTION:

With:

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

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.4-1, except for main steam line isolation valves* and valves which are hydrostatically leak tested per Table 3.6.4-1, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line through the isolation valve, or

*Exemption to Appendix J of 10 CFR 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. The combined leakage rate for all penetrations shown in Table 3.6.4-1 as secondary containment bypass leakage paths exceeding 30,000 cc/hr or

e. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 3 gpm,

restore:

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

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.4-1, except for main steam line isolation valves* and valves which are hydrostatically leak tested per Table 3.6.4-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than 11.5 scf per hour for any one main steam line through the isolation valve, and

d. The combined leakage rate for all penetrations shown in Table 3.6.4-1 as secondary containment bypass leakage paths to less than or equal to 30,000 cc/hr and

e. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 3 gpm,

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4 - (1972):

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 $\pm$ 10 month intervals during shutdown at P_a, 6.31 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.

*Exception to Appendix J of 10 CFR 50.

SURVEILLANCE REQUIREMENTS (Continued)

b. 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.

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

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

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

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 6.31 psig.

d. Type B and C tests shall be conducted with gas at P_a, 6.31 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.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

g. Type B periodic tests are not required for penetrations continuously monitored by the Primary Containment Penetration Pressurization System, provided the system is OPERABLE per Specification 3.6.1.9.

h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, 6.31 psig, at intervals no greater than once per 3 years.

i. 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, 6.94 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

j. 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.

k. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.3. and 4.6.1.8.4.

l. The provisions of Specification 4.0.2 are not applicable to 24 month or 40 ± 10 month surveillance intervals.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.4 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 6.31 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one containment air lock door inoperable:

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

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

3. Otherwise, be in at least HOT 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 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.

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

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to Pa, 6.31 psig.

b. By conducting an overall air lock leakage test at P_a, 6.31 psig, and verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months#,

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

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

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

1. Demonstrating (two) seal air flask pressure instrumentation channel(s) 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,

with a low pressure setpoint of $\geq$80 psig.

2. At least once per 7 days, verifying seal air flask pressure to be greater than or equal to 80 psig.

3. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 4.2 psig from 110 psig within 48 hours.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

CONTAINMENT SYSTEMS

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.5 Two independent main steam positive leakage control system (MSPLCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSPLCS 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

4.6.1.5 Each MSPLCS subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1. Blower OPERABILITY by starting the injection system from the control room and operating the system for at least (15) minutes through the drain line.

   2. The (functional availability) of the MSPLCS by ______ ______________________________.

   3. ______________________________.

b. During each COLD SHUTDOWN (, if not performed within the previous 92 days, by cycling each remote, manual and automatic motor operated valve through at least one complete cycle of full travel) (in accordance with Specification 4.0.5).

c. At least once per 18 months by performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that greater than or equal to (   ) psig sealing pressure is established in each steam line.

d. By verifying the (flow, pressure, temperature and level) (operating) instrumentation to be OPERABLE by performance of a:

   1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
   2. CHANNEL CALIBRATION at least once per 18 months.

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of 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.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

CONTAINMENT SYSTEMS

CONTAINMENT INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Containment to secondary containment differential pressure shall be maintained between - 0.1 and + 1.5 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the containment to secondary containment differential pressure outside of the specified limits, restore the differential pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The containment to secondary containment differential pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.8 Containment average air temperature shall not exceed 90°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the containment average air temperature greater than 90°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

	Elevation	Azimuth
a.	________	________
b.	________	________
c.	________	________
d.	________	________
e.	________	________
f.	________	________

CONTAINMENT SYSTEMS

DRYWELL AND CONTAINMENT PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 The containment purge 36 inch supply and exhaust isolation valves shall be OPERABLE and:

a. Each 36 inch purge valve shall be sealed closed.

b. Each 36 inch purge valve may be open for purge system operation with such operation limited to 90 hours per 365 days for reducing airborne activity and pressure control.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a 36 inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal the 36 inch valve(s) or otherwise isolate the penetration within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the 36 inch containment purge system in operation and/or with the 36 inch supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, discontinue 36 inch purge system operation and close the open 36 inch valve(s) or otherwise isolate the penetration(s) within four hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.3 and/or 4.6.1.8.4, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

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

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.9.3 At least once per 92 days each 36 inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to (0.01) L_a when pressurized to P_a.

CONTAINMENT SYSTEMS

PENETRATION VALVE LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one PVLCS division inoperable, restore the inoperable division to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.10 Each PVLCS division shall be demonstrated OPERABLE:

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

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each remote, manual and automatic motor operated valve through at least one complete cycle of full travel in accordance with Specification 4.0.5.

c. At least once per 18 months by:

   1. Performance of a functional test which includes simulated actuation of the system throughout its operating sequence, and verifying that each automatic valve actuates to its correct position and that a sealing pressure greater than or equal to 21 psig is established in each sealing valve, and

   2. Verifying a total sealing air leakage rate into the primary containment at a test pressure of 43 psid for:

      a) Division 1 of $\leq$ 209 scfh, and
      b) Division 2 of $\leq$ 209 scfh.

   3. Leakage from valves equipped with the PVLCS will be included in computation of 0.6 L_a.

d. By verifying the flow, pressure, temperature and level operating instrumentation to be OPERABLE by performance of a:

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

   2. CHANNEL CALIBRATION at least once per 18 months.

3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. After each closing of the combination equipment hatch/personnel door, if opened following the drywell bypass leakage rate test, by verifying that the combination equipment hatch/personnel door is in place and by leak rate testing the gap between the seals and verifying that the measured leakage rate for these seals, when pressurized with gas at 3.0 psig:

   1. If the personnel door was opened, is less than or equal to 200 cc per hour, and

   2. If the equipment hatch was opened, is less than or equal to 75 oc per hour.

b. At least once per 31 days by verifying that all drywell penetrations** not capable of being closed by OPERABLE drywell automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.4-1 of Specification 3.6.4.

c. By verifying the drywell air lock OPERABLE per Specification 3.6.2.3.

*See Special Test Exception 3.10.1.

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

SURVEILLANCE REQUIREMENTS (Continued)

d. By verifying the suppression pool OPERABLE per Specification 3.6.3.1.

e. By verifying the personnel door inflatable seal system OPERABLE by:

   1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 65 psig.

   2. At least once per 18 months conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 104 psig within 48 hours.

CONTAINMENT SYSTEMS

DRYWELL BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.2.2 Drywell bypass leakage shall be less than or equal to 10% of the minimum acceptable $A/\sqrt{k}$ design value of 1.0 $ft.^2$

APPLICABILITY: When DRYWELL INTEGRITY is required per Specification 3.6.2.1.

ACTION:

With the drywell bypass leakage greater than 10% of the minimum acceptable $A/\sqrt{k}$ design value of 1.0 $ft.^2$, restore the drywell bypass leakage to within the limit prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The drywell bypass leakage rate test shall be conducted at least once per 18 months at an initial differential pressure of 3.0 psi and the $A/\sqrt{k}$ shall be calculated from the measured leakage. One drywell airlock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other leakage rate test.

a. If any drywell bypass leakage test fails to meet the specified limit, the schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the limit, a test shall be performed at least every 9 months until two consecutive tests meet the limit, at which time the 18 month test schedule may be resumed.

b. The provisions of Specification 4.0.2 are not applicable.

DRYWELL AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.2.3 The drywell 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 drywell, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 2 scf per hour at Pa, 15.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one drywell air lock door inoperable:

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

    2. Operation may then continue 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 drywell 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.

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

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The drywell air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 2 scf per hour when the gap between the door seals is pressurized to Pa, 15.0 psig.

b. By conducting an overall air lock leakage test at P_a, 15.0 psig and verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months[#].

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

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

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

   1. Demonstrating two seal air flask pressure instrumentation channels OPERABLE with a low pressure setpoint of $\geq$80 psig by performance of a:

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

      b) CHANNEL CALIBRATION at least once per 18 months.

   2. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 80 psig.

   3. At least once per 18 months conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 3 psig from 110 psig within 48 hours.

[#]The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

DRYWELL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.4 The structural integrity of the drywell shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.2.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.6.2.4.2 Reports Any abnormal degradation of the drywell structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

DRYWELL INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.2.5 Drywell to containment differential pressure shall be maintained between - 0.1 and + 1.5 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell to containment differential pressure outside of the specified limits, restore the differential pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.5 The drywell to containment differential pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.2.6 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

4.6.2.6 The drywell average air temperature shall be the (arithmetical) average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	______	______
b.	______	______
c.	______	______
d.	______	______
e.	______	______
f.	______	______

CONTAINMENT SYSTEMS

DRYWELL VENT AND PURGE

LIMITING CONDITION FOR OPERATION

3.6.2.7 The drywell vent and purge system supply and exhaust valves shall be closed whenever the 36 inch containment purge system supply or exhaust valves are open; except while in OPERATIONAL CONDITION 3, the drywell vent and purge system 24 inch valves may be open during operation of the drywell vent and purge mode of the containment cooling system for up to 90 hours per 365 days for the purpose of reducing drywell airborne radioactivity levels prior to and during personnel entries or for controlling drywell pressure. The drywell may be vented for up to 5 hours per 365 days in OPERATIONAL CONDITIONS 1 and 2, for controlling drywell pressure by opening the 24 inch drywell purge supply or exhaust valves; however, only one line may e open at a time.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the drywell vent and purge system supply or exhaust valves open in OPERATIONAL CONDITIONS 1 and 2 and the 36 inch containment purge system supply or exhaut valves open, immediately close the drywell vent and purge system valves or be in at least HOT SHUTDOWN within the next 12 hous.

b. With the drywell vent and purge system supply or exhaust valves open during OPERATIONAL CONDITIONS 1 and 2 for more than 5 hours per 365 days, immediately close the drywell vent valves or be in at least HOT SHUTDOWN within the next 12 hours.

c. With both the drywell purge supply and exhaust valves open at the same time in OPERATIONAL CONDITIONS 1 and 2, immediately isolate either the supply or exhaust line; otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

d. With the drywell vent or purge mode of the containment cooling system in operation during OPERATIONAL CONDITION 3 for more than 90 hours per 365 days, immediately close the drywell vent and purge 24 inch valves or be in at least COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.7 At least once per 7 days, determine the cumultive time that:

1. The drywell vent and purge system supply or exhaust valves have been open during OPERATIONAL CONDITIONS 1 and 2 during the past 365 days, and

2. The drywell vent and purge mode of the containment cooling system has been in operation during OPERATIONAL CONDITION 3 within the past 365 days.

CONTAINMENT SYSTEMS

3/4.6.3 DEPRESSURIZATON SYSTEMS

SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.6.3.1 The suppression pool shall be OPERABLE with the pool water:

a. Volume between 137,571 ft^3 and 141,036 ft^3, equivalent to a level between 19'6" and 20'0" and a

b. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

   1. 105°F during testing which adds heat to the suppression pool.

   2. 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

   3. 120°F wth the main steam line isolation valves closed following a scram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

b. In OPERATIONAL CONDITION 1 or 2 with the suppression pool average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

   1. With the suppression pool average water temperature greater than 105°F during testing which adds heat to the suppression pool, stop all testing which adds heat to the suppression pool and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2. With the suppression pool average water temperature greater than:

   a) 95°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

   b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

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

c. With one suppression pool water temperature instrumentation channel in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression pool water temperature to be within the limits at least once per 12 hours.

d. With both suppression pool water temperature instrumentation channels in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore at least one inoperable water temperature instrumentation channel in each pair of temperature instrumentation channels in the same sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The suppression pool shall be demonstrated OPERABLE:

a. By verifying the suppression pool 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 pool average water temperature to be less than or equal to 95°F, except:

   1. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature less than or equal to 105°F.

SURVEILLANCE REQUIREMENTS (Continued)

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

   a) Suppression pool 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 pool average water temperature has exceeded 95°F for more than 24 hours.

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

c. By verifying (at least) fourteen suppression pool water temperature instrumentation channels, at least two channels in each suppression pool sector, 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 ≤ ( )°F.

CONTAINMENT SYSTEMS

CONTAINMENT UNIT COOLERS

LIMITING CONDITION FOR OPERATION

3.6.3.2 Both containment unit coolers shall be OPERABLE and capable of rejecting heat to the Standby Service Water System.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the containment unit coolers inoperable, restore the inoperable unit cooler to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With both containment unit coolers inoperable, restore at least one unit cooler to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.2 Both containment unit coolers shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each pressure relief and backdraft damper in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

b. By verifying that each of the required unit cooler develops a flow of at least 50000 cfm on recirculation flow through the unit cooler at least once per 92 days.

c. At least once per 18 months by performance of a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each pressure relief and backdraft damper in the flow path actuates to its correct position.

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR pump; and

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. By verifying that each of the required RHR pumps develops a flow of at least 5050 gpm on recirculation flow through the RHR heat exchangers to the suppression pool when tested pursuant to Specification 4.0.5.

*Whenever both RHR 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.

3/4.6.4 CONTAINMENT AND DRYWELL ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 The containment and drywell isolation valves in Table 3.6.4-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.4-1.

APPLICABILITY: As shown in Table 3.6.4-1.

ACTION:

a. With one or more of the containment or drywell isolation valves shown in Table 3.6.4-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   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 blind 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 all operations involving CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

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

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.4.2 Each automatic isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on an isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.4.3 The isolation time of each power operated or automatic valve shown in Table 3.6.4-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.4.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow at greater than a (10) psid differential pressure.

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves							
1. Primary Containment							
1B21*	AOV F022A(b)(g)	1KJB*Z1A	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F022B(b)(g)	1KJB*Z1B	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F022C(b)(g)	1KJB*Z1C	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F022D(b)(g)	1KJB*Z1D	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F028A(g)	1KJB*Z1A	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F028B(g)	1KJB*Z1B	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F028C(g)	1KJB*Z1C	6	1, 2, 3 and ##	5	No	6.31
1B21*	AOV F028D(g)	1KJB*Z1D	6	1, 2, 3 and ##	5	No	6.31
1B21*	MOVF067A(g)	1KJB*Z1A	6	1, 2, 3 and ##	7.5	No	6.31
1B21*	MOVF067B(g)	1KJB*Z1B	6	1, 2, 3 and ##	7.5	No	6.31
1B21*	MOVF067C(g)	1KJB*Z1C	6	1, 2, 3 and ##	7.5	No	6.31
1B21*	MOVF067D(g)	1KJB*Z1D	6	1, 2, 3 and ##	7.5	No	6.31
1B21*	MOVF016(b)(g)	1KJB*Z2	6	1, 2, 3 and ##	16.5	No	6.31
1B21*	MOVF019(g)	1KJB*Z2	6	1, 2, 3 and ##	17.6	No	6.31
1E12*	MOVF053A	1KJB*Z3A	5	1, 2, 3 and ##	18.7	No	6.31
1E12*	MOVF053B	1KJB*Z3B	5	1, 2, 3 and ##	18.7	No	6.31
1E12*	MOVF023(b)	1KJB*Z19, IDRB#Z13	5	1, 2, 3 and ##	36.3	No	6.31
1E12*	MOVF0008	1KJB*Z20	5	1, 2, 3 and ##	29.7	No	6.31
1E12*	MOVF009(b)	1KJB*Z20	5	1, 2, 3 and ##	25.3	No	6.31
1E12*	MOVF037A	1KJB*Z21A	5	1, 2, 3 and ##	73.7	No	6.31
1E12*	MOVF037B	1KJB*Z21B	5	1, 2, 3 and ##	74.8	No	6.31
1E33*	MOVF008(d)	1KJB*Z1A,B,C,D	4	1, 2, 3 and ##	N/A	No	6.31

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves							
1. Primary Containment (Continued)							
1G33*	MOVF028	1KJB*Z4	7	1, 2, 3 and ##	20.9	Yes	6.31
1G33*	MOVF040	1KJB*Z6	7	1, 2, 3 and ##	24.2	No	6.31
1G33*	MOVF001(b)(j)	1KJB*Z7	7	1, 2, 3 and ##	19.8	No	6.31
1G33*	MOVF053	1KJB*Z129	7	1, 2, 3 and ##	5.5	No	6.31
1G33*	MOVF034(h)	1KJB*Z4	7	1, 2, 3 and ##	10.9	Yes	6.31
1G33*	MOVF039(h)	1KJB*Z6	7	1, 2, 3 and ##	24.2	No	6.31
1G33*	MOVF004(h)(j)	1KJB*Z7	7	1, 2, 3 and ##	6.6	No	6.31
1G33*	MOVF054(h)	1KJB*Z129	7	1, 2, 3 and ##	5.5	No	6.31
1WCS*	MOV178	1KJB*Z5	1	1, 2, 3 and ##	12.1	Yes	6.31
1WCS*	MOV172	1KJB*Z5	1	1, 2, 3 and ##	12.6	Yes	6.31
1E22*	MOVF023	1KJB*Z11	1	1, 2, 3 and ##	50	No	6.31
1E12*	MOVF024A	1KJB*Z24A	10	1, 2, 3 and ##	63.8	No	6.31
1E12*	MOVF011A	1KJB*Z24A	10	1, 2, 3 and ##	34.1	No	6.31
1E21*	MOVF012	1KJB*Z24A	10	1, 2, 3 and ##	57.2	No	6.31
1E12*	MOVF024B	1KJB*Z24B	10	1, 2, 3 and ##	63.8	No	6.31
1E12*	MOVF011B	1KJB*Z24B	10	1, 2, 3 and ##	30.8	No	6.31
1E12*	MOVF021	1KJB*24C	10	1, 2, 3 and ##	97.9	No	6.31
1SFC*	MOV119	1KJBZ26	1	1, 2, 3 and ##	68	No	6.31
1SFC*	MOV120	1KJBZ27	1	1, 2, 3 and ##	62.7	No	6.31
1SFC*	MOV122	1KJBZ27	1	1, 2, 3 and ##	63.8	No	6.31
1SFC*	MOV139	1KJBZ28	1	1, 2, 3 and ##	39.6	No	6.31
1SFC*	MOV121	1KJBZ28	1	1, 2, 3 and ##	39.6	No	6.31

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves						
1. Primary Containment (Continued)						
1DFR* AOV102(b)	1KJB*Z35	1	1, 2, 3 and ##	N/A	No	6.31
1DRF* AOV101(b)	1KJB*Z35	1	1, 2, 3 and ##	N/A	No	6.31
1DER* AOV127(b)	1KJB*Z38	1	1, 2, 3 and ##	N/A	No	6.31
1DER* AOV126(b)	1KJB*Z38	1	1, 2, 3 and ##	N/A	No	6.31
1FPW* MOV121	1KJB*Z41	1	1, 2, 3 and ##	34.1	Yes	6.31
1SAS* MOV102	1KJB*Z44	1	1, 2, 3 and ##	22.0	Yes	6.31
1IAS* MOV106	1KJB*Z46	1	1, 2, 3 and ##	18.7	Yes	6.31
1CCP* MOV138	1KJB*Z48	1	1, 2, 3 and ##	22.0	No	6.31
1CCP* MOV158	1KJB*Z49	1	1, 2, 3 and ##	23.1	No	6.31
1CCP* MOV159	1KJB*Z49	1	1, 2, 3 and ##	24.2	No	6.31
1SWP* MOV5A	1KJB*Z53A	1	1, 2, 3 and ##	50.6	No	6.31
1SWP* MOV5B	1KJB*Z53B	1	1, 2, 3 and ##	53.9	No	6.31
1HVN* MOV102	1KJB*Z131	1	1, 2, 3 and ##	31.9	Yes	6.31
1HVN* MOV128	1KJB*Z131	1	1, 2, 3 and ##	28.6	Yes	6.31
1HVN* MOV127	1KJB*Z132	1	1, 2, 3 and ##	27.5	Yes	6.31
1CNS* MOV125	1KJB*Z134	1	1, 2, 3 and ##	22.0	Yes	6.31

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves							
1. Primary Containment (Continued)							
1E51*	MOVF063(b)	1KJB*Z15	2	1, 2, 3 and ##	9.9	No	6.31
1E51*	MOVF076(b)	1KJB*Z15	2	1, 2, 3 and ##	13.4	No	6.31
1E51*	MOVF064	1KJB*Z15	2	1, 2, 3 and ##	9.9	No	6.31
1E51*	MOVF031	1KJB*Z16	2	1, 2, 3 and ##	21.8	No	6.31
1E51*	MOVF077	1KJB*Z17	3	1, 2, 3 and ##	14.2	No	6.31
1E51*	MOVF078	1KJB*Z18B,C	3	1, 2, 3 and ##	16.5	No	6.31
1HVR*	AOV165	1KJB*Z31	8	1, 2, 3 and ##	3	No	6.31
1HVR*	AOV123	1KJB*Z31	8	1, 2, 3 and ##	3	No	6.31
1HVR*	AOV128	1KJB*Z33	8	1, 2, 3 and ##	3	No	6.31
1HVR*	AOV166	1KJB*Z33	8	1, 2, 3 and ##	3	No	6.31
1SSR*	SOV130	1KJB*2601B	10	1, 2, 3 and ##	3	No	6.31
1SSR*	SOV131	1KJB*2601B	10	1, 2, 3 and ##	3	No	6.31

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves							
2. Drywell							
1HVR*	AOV147	1DRB*Z32	1	1, 2, 3	3	No	N/A
1CCP*	MOV142	1DRB*Z50	1	1, 2, 3	30	No	N/A
1CCP*	MOV144	1DRB*Z51	1	1, 2, 3	30	No	N/A
1CCP*	MOV143	1DRB*Z51	1	1, 2, 3	30	No	N/A
1SWP*	MOV4A	1DRB*Z54	1	1, 2, 3	52.8	No	N/A
1SWP*	MOV4B	1DRB*Z54	1	1, 2, 3	51.7	No	N/A
1SWP*	MOV5A	1DRB*Z55	1	1, 2, 3	50.6	No	N/A
1SWP*	MOV5B	1DRB*Z55	1	1, 2, 3	53.9	No	N/A
1RCS*	MOV58A	1DRB*Z152	1	1, 2, 3	11.0	No	N/A
1RCS*	MOV59A	1DRB*Z153	1	1, 2, 3	10.6	No	N/A
1RCS*	MOV60A	1DRB*Z154	1	1, 2, 3	6.3	No	N/A
1RCS*	MOV61A	1DRB*Z155	1	1, 2, 3	8.6	No	N/A
1RCS*	MOV58B	1DRB*Z156	1	1, 2, 3	10.6	No	N/A
1RCS*	MOV59B	1DRB*Z157	1	1, 2, 3	10.8	No	N/A
1RCS*	MOV60B	1DRB*Z158	1	1, 2, 3	6.38	No	N/A
1RCS*	MOV61B	1DRB*Z159	1	1, 2, 3	8.9	No	N/A
1HVR*	AOV125	1DRB*Z32	1	1, 2, 3	3	No	N/A
1HVR*	AOV126	1DRB*Z34	1	1, 2, 3	3	No	N/A
1HVR*	AOV148	1DRB*Z34	1	1, 2, 3	3	No	N/A

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP	(APPLICABLE OPERATIONAL CONDITIONS)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)
a. Automatic Isolation Valves							
2. Drywell (Continued)							
1CPM*	MOV2A	1DRBZ57A	10	1, 2, 3	30	No	N/A
1CPM*	MOV4A	1DRBZ57A	10	1, 2, 3	30	No	N/A
1CPM*	MOV2B	1DRBZ57B	10	1, 2, 3	30	No	N/A
1CPM*	MOV4B	1DRBZ57B	10	1, 2, 3	30	No	N/A
1CPM*	MOV3A	1DRBZ58A	10	1, 2, 3	30	No	N/A
1CPM*	MOV1A	1DRBZ58A	10	1, 2, 3	30	No	N/A
1CPM*	MOV3B	1DRBZ58B	10	1, 2, 3	30	No	N/A
1CPM*	MOV1B	1DRBZ58B	10	1, 2, 3	30	No	N/A
1B33*	AOVF019	1DRBZ449	9	1, 2, 3	(6)	No	N/A
1B33*	AOVF020	1DRBZ449	9	1, 2, 3	(6)	No	N/A

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
b. Manual Isolation Valves					
1. Primary Containment					
1E12*	F099A	1KJB*Z21A	No	6.31	1, 2, 3 and ##
1E12*	F099B	1KJB*Z21B	No	6.31	1, 2, 3 and ##
1HVR*	V8	1KJB*Z602A	No	6.31	1, 2, 3 and ##
1HVR*	V10	1KJB*Z602B	No	6.31	1, 2, 3 and ##
1LSV*	V64	1KJB*Z602D	No	6.31	1, 2, 3 and ##
1HVR*	V12	1KJB*Z602F	No	6.31	1, 2, 3 and ##
1LMS*	V14	1KJB*Z603A	No	6.31	1, 2, 3 and ##
1LMS*	V12	1KJB*Z603A	No	6.31	1, 2, 3 and ##
1LMS*	V7	1KJB*Z603C	No	6.31	1, 2, 3 and ##
1LMS*	V16	1KJB*Z603C	No	6.31	1, 2, 3 and ##
1CMS*	V2	1KJB*Z605A	No	6.31	1, 2, 3 and ##
1CMS*	V3	1KJB*Z605B	No	6.31	1, 2, 3 and ##
1HVR*	V14	1KJB*Z606A	No	6.31	1, 2, 3 and ##
1HVR*	V16	1KJB*Z606B	No	6.31	1, 2, 3 and ##
1CMS*	V16	1KJB*Z606C	No	6.31	1, 2, 3 and ##
1CMS*	V15	1KJB*Z606D	No	6.31	1, 2, 3 and ##
1LSV*	V65	1KJB*Z606E	No	6.31	1, 2, 3 and ##
1HVR*	V18	1KJB*Z606F	No	6.31	1, 2, 3 and ##
1E12*	VF044A	1KJB*Z21A	No	6.31	1, 2, 3 and ##
1E12*	VF044B	1KJB*Z21B	No	6.31	1, 2, 3 and ##

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
b.	Manual Isolation Valves				
1.	Primary Containment (Continued)				
1FWS*	MOV7A(e)	1KJB*Z3A	Yes	6.31	1, 2, 3 and ##
1FWS*	MOV7B(e)	1KJB*Z3B	Yes	6.31	1, 2, 3 and ##
1E22*	MOVF015(e)	1KJB*Z8	No	6.31	1, 2, 3 and ##
1E22*	MOVF004(b)(e)	1KJB*Z9,IDRB#Z10	No	6.31	1, 2, 3 and ##
1E22*	MOVF012(e)	1KJB*Z11	No	6.31	1, 2, 3 and ##
1E21*	MOVF001(e)	1KJB*Z12	No	6.31	1, 2, 3 and ##
1E21*	MOVF005(b)(e)	1KJB*Z13 IDRB#Z14	No	6.31	1, 2, 3 and ##
1E51*	MOVF068(e)	1KJB*Z17	No	6.31	1, 2, 3 and ##
1E51*	MOVF019(e)	1KJB*Z18A	No	6.31	1, 2, 3 and ##
1E51*	MOVF013(b)(e)	1KJB*Z119, IDRB*Z130	No	6.31	1, 2, 3 and ##
1E12*	MOVF027A(e)	1KJB*Z21A	No	6.31	1, 2, 3 and ##
1E12*	MOVF042A(e)	1KJB*Z21A	No	6.31	1, 2, 3 and ##
1E12*	MOVF027B(e)	1KJB*Z21B	No	6.31	1, 2, 3 and ##
1E12*	MOVF042B(e)	1KJB*Z21B	No	6.31	1, 2, 3 and ##
1E12*	MOVF042C(e)	1KJB*Z21C	No	6.31	1, 2, 3 and ##
1E12*	MOVF073A(e)	1KJB*Z23A	No	6.31	1, 2, 3 and ##
1E12*	MOVF073B(e)	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1E12*	MOVF064A(e)	1KJB*Z24A	No	6.31	1, 2, 3 and ##
1E21*	MOVF011(e)	1KJB*Z24A	No	6.31	1, 2, 3 and ##
1E12*	MOVF064B(e)	1KJB*Z24B	No	6.31	1, 2, 3 and ##
1E12*	MOVF064C(e)	1KJB*Z24C	No	6.31	1, 2, 3 and ##
1E12*	MOVF004A(e)	1KJB*Z25B	No	6.31	1, 2, 3 and ##
1E12*	MOVF105(e)	1KJB*Z25C	No	6.31	1, 2, 3 and ##

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

b. Manual Isolation Valves

1. Primary Containment (Continued)

VALVE NUMBER		PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
1C11*	MOVF083(e)	1KJB*Z29	No	6.31	1, 2, 3 and ##
1CPP*	MOV104(e)	1KJB*Z33	No	6.31	1, 2, 3 and ##
1CPP*	MOV105(e)	1KJB*Z33	No	6.31	1, 2, 3 and ##
1SWP*	MOV507A(e)	1KJB*Z52A	No	6.31	1, 2, 3 and ##
1SWP*	MOV507B(e)	1KJB*Z52B	No	6.31	1, 2, 3 and ##
1SWP*	MOV81A(e)	1KJB*Z53A	No	6.31	1, 2, 3 and ##
1SWP*	MOV81B(e)	1KJB*Z53B	No	6.31	1, 2, 3 and ##
1SWP*	MOV503A(e)	1KJB*Z53A	No	6.31	1, 2, 3 and ##
1SWP*	MOV503B(e)	1KJB*Z53B	No	6.31	1, 2, 3 and ##
1SVV*	MOV1B(e)	1KJB*Z102	No	6.31	1, 2, 3 and ##
1SVV*	MOV1A(e)	1KJB*Z103	No	6.31	1, 2, 3 and ##
1CPP*	SOV140(e)	1KJB*Z31	No	6.31	1, 2, 3 and ##
1CMS*	SOV35D(e)	1KJB*Z601E	No	6.31	1, 2, 3 and ##
1CMS*	SOV31B(e)	1KJB*Z601E	No	6.31	1, 2, 3 and ##
1CMS*	SOV35B(e)	1KJB*Z601F	No	6.31	1, 2, 3 and ##
1CMS*	SOV31D(e)	1KJB*Z601F	No	6.31	1, 2, 3 and ##
1CMS*	SOV35C(e)	1KJB*Z605E	No	6.31	1, 2, 3 and ##
1CMS*	SOV31A(e)	1KJB*Z605E	No	6.31	1, 2, 3 and ##
1CMS*	SOV35A(e)	1KJB*Z605F	No	6.31	1, 2, 3 and ##
1CMS*	SOV31C(e)	1KJB*Z605F	No	6.31	1, 2, 3 and ##

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

b. Manual Isolation Valves

2. Drywell

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
1SAS* V489	1DRB*Z45	No	N/A	1, 2, 3
1IAS* V79	1DRB*Z47	No	N/A	1, 2, 3
1HVN* V542	1DRB*Z54	No	N/A	1, 2, 3
1SWP* V205	1DRB*Z54	No	N/A	1, 2, 3
1SWP* V206	1DRB*Z55	No	N/A	1, 2, 3
1SVV* V50	1DRB*Z107	No	N/A	1, 2, 3
1SVV* V53	1DRB*Z112	No	N/A	1, 2, 3
1IAS* V237	1DRB*Z166	No	N/A	1, 2, 3
1IAS* V238	1DRB*Z166	No	N/A	1, 2, 3
1RCS* V132	1DRB*Z152	No	N/A	1, 2, 3
1RCS* V131	1DRB*Z153	No	N/A	1, 2, 3
1RCS* V162	1DRB*Z154	No	N/A	1, 2, 3
1RCS* V156	1DRB*Z155	No	N/A	1, 2, 3
1RCS* V187	1DRB*Z156	No	N/A	1, 2, 3
1RCS* V186	1DRB*Z157	No	N/A	1, 2, 3
1RCS* V217	1DRB*Z158	No	N/A	1, 2, 3
1RCS* V211	1DRB*Z159	No	N/A	1, 2, 3
1B21* MOVF005(e)	1DRB*Z161	No	N/A	1, 2, 3
1CMS* SOV34F(e)	1DRB*Z164	No	N/A	1, 2, 3
1CMS* SOV34E(e)	1DRB*Z165	No	N/A	1, 2, 3

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

	VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c.	Other Isolation Valves				
	1. Primary Containment				
1B21*	AOVF032A[c]	1KJB*Z3A	Yes	6.94	1, 2, 3 and ##
1B21*	AOVVF010A[b]	1KJB*Z3A	Yes	6.94	1, 2, 3 and ##
1B21*	AOVF032B[c]	1KJB*Z3B	Yes	6.94	1, 2, 3 and ##
1B21*	AOVVF010B[b]	1KJB*Z3B	Yes	6.94	1, 2, 3 and ##
1E22*	AOVF005[b][c]	1KJB*Z9, IDRB#Z10	No	6.31	1, 2, 3 and ##
1E22*	RVF014	1KJB*Z11	No	6.31[a]	1, 2, 3 and ##
1E22*	RVF035	1KJB*Z11	No	6.31[a]	1, 2, 3 and ##
1E22*	RVF039	1KJB*Z11	No	6.31[a]	1, 2, 3 and ##
1E21*	AOVF006[b][c]	1KJB*Z13,IDRB*Z14	No	6.31	1, 2, 3 and ##
1E51*	AOVF065[b][c]	1KJB*Z19,IDRB*Z130	No	6.31	1, 2, 3 and ##
1E51*	AOVF066[b][c]	1KJB*Z19,IDRB*Z130	No	6.31	1, 2, 3 and ##
1E12*	AOVF041C[b][c]	1KJB*Z21C,IDRB#Z22C	No	6.31	1, 2, 3 and ##
1RHS*	RV3A	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1E12*	RVF055A	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1E12*	RVF025A	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1E12*	RVF017A	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1E12*	RVF005	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1E21*	RVF018	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1E21*	RVF031	1KJB*Z23A	No	6.31[a]	1, 2, 3 and ##
1RHS*	RV38	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1E12*	RVF055B	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1E12*	RVF025C	1KJB*Z23B	No	6.31	1, 2, 3 and ##

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

c. Other Isolation Valves

1. Primary Containment (Continued)

VALVE NUMBER		PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
1E12*	RVF025B	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1E12*	RVF030	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1E12*	RVF101	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1E12*	RVF017B	1KJB*Z23B	No	6.31	1, 2, 3 and ##
1SFC*	V101	1KJB*Z26	No	6.31	1, 2, 3 and ##
1C11*	VF122	1KJB*Z29	No	6.31	1, 2, 3 and ##
1DFR*	V180	1KJB*Z35	No	6.31	1, 2, 3 and ##
1FPW*	V263	1KJB*Z41	Yes	6.31	1, 2, 3 and ##
1SAS*	V486	1KJB*Z44	Yes	6.31	1, 2, 3 and ##
1IAS*	V80	1KJB*Z46	Yes	6.31	1, 2, 3 and ##
1CCP*	V118	1KJB*Z48	No	6.31	1, 2, 3 and ##
1SWP*	V174	1KJB*Z52A	No	6.31	1, 2, 3 and ##
1SWP*	V175	1KJB*Z52B	No	6.31	1, 2, 3 and ##
1SVV*	V9	1KJB*Z102	No	6.31	1, 2, 3 and ##
1SVV*	V31	1KJB*Z103	No	6.31	1, 2, 3 and ##
1HVN*	V541	1KJB*Z132	Yes	6.31	1, 2, 3 and ##
1CNS*	V86	1KJB*Z134	Yes	6.31	1, 2, 3 and ##

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

c. Other Isolation Valves

2. Drywell

VALVE NUMBER		PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
1B21*	RVF047A	1DRB*Z136	No	N/A	1, 2, 3
1B21*	RVF041A	1DRB*Z137	No	N/A	1, 2, 3
1B21*	RVF051G	1DRB*Z138	No	N/A	1, 2, 3
1B21*	RVF041L	1DRB*Z139	No	N/A	1, 2, 3
1B21*	RVF047C	1DRB*Z140	No	N/A	1, 2, 3
1B21*	RVF041G	1DRB*Z141	No	N/A	1, 2, 3
1B21*	RVF051C	1DRB*Z142	No	N/A	1, 2, 3
1B21*	RVF041C	1DRB*Z143	No	N/A	1, 2, 3
1B21*	RVF047B	1DRB*Z144	No	N/A	1, 2, 3
1B21*	RVF041B	1DRB*Z145	No	N/A	1, 2, 3
1B21*	RVF051B	1DRB*Z146	No	N/A	1, 2, 3
1B21*	RVF041F	1DRB*Z147	No	N/A	1, 2, 3
1B21*	RVF047F	1DRB*Z148	No	N/A	1, 2, 3
1B21*	RVF041D	1DRB*Z149	No	N/A	1, 2, 3
1B21*	RVF047D	1DRB*Z150	No	N/A	1, 2, 3
1B21*	RVF051D	1DRB*Z151	No	N/A	1, 2, 3
1E12*	AOVF041A(c)	1DRB*Z22A	No	N/A	1, 2, 3
1E12*	AOVF041B(c)	1DRB*Z22B	No	N/A	1, 2, 3

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c. Other Isolation Valves				
2. Drywell (Continued)				
1DFR* V4	1DRB*Z37A	No	N/A	1, 2, 3
1DFR* V3	1DRB*Z37A	No	N/A	1, 2, 3
1DFR* V1	1DRB*Z37B	No	N/A	1, 2, 3
1DFR* V2	1DRB*Z37B	No	N/A	1, 2, 3
1DER* V14	1DRB*Z40A	No	N/A	1, 2, 3
1DER* V15	1DRB*Z40A	No	N/A	1, 2, 3
1DER* V16	1DRB*Z40B	No	N/A	1, 2, 3
1DER* V17	1DRB*Z40B	No	N/A	1, 2, 3
1SAS* V487	1DRB*Z45	No	N/A	1, 2, 3
1IAS* V78	1DRB*Z47	No	N/A	1, 2, 3
1CCP* V119	1DRB*Z50	No	N/A	1, 2, 3
1SWP* RV119	1DRB*Z54	No	N/A	1, 2, 3
1CAI* VEXF004A	1DRB*Z56	No	N/A	1, 2, 3
1CAI* VEXF004B	1DRB*Z56	No	N/A	1, 2, 3
1CAI* VF006	1DRB*Z56	No	N/A	1, 2, 3
1CAI* VF007	1DRB*Z56	No	N/A	1, 2, 3
1CCP* V133	1DRB*Z51	No	N/A	1, 2, 3

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

VALVE NUMBER		PENETRATION NUMBER	SECONDARY CONTAINMENT BYPASS PATH(f) (Yes/No)	TEST PRESSURE (psig)	(APPLICABLE OPERATIONAL CONDITIONS)
c.	Other Isolation Valves				
2.	Drywell (Continued)				
1B21*	VF036A	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF036F	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF036G	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF036P	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF039C	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF039H	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF039K	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF039S	1DRB*Z107	No	N/A	1, 2, 3
1B21*	VF036J	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF036L	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF036M	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF036N	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF036R	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF039B	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF039D	1DRB*Z112	No	N/A	1, 2, 3
1B21*	VF039E	1DRB*Z112	No	N/A	1, 2, 3
1B33*	VF013A	1DRB*Z133	No	N/A	1, 2, 3
1B33*	VF017A	1DRB*Z133	No	N/A	1, 2, 3
1B33*	VF013B	1DRB*Z135	No	N/A	1, 2, 3
1B33*	VF017B	1DRB*Z135	No	N/A	1, 2, 3

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

NOTES

(a) Not subject to Type C leakage tests. Valve(s) will be indicated in the type A test.

(b) Also isolates the drywell

(c) TESTABLE CHECK VALVE

(d) OPENS ON ISOLATION SIGNAL

(e) RECEIVES A REMOTE MANUAL ISOLATION SIGNAL

(f) THIS LINE IS SEALED BY THE PENETRATION VALVE LEAKAGE CONTROL SYSTEM

(g) This valve sealed by the main steam positive leakage control system (MS-PLCS)

(h) ALSO ISOLATES ON HIGH NONREGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE (RWCU)

(j) VALVES G33* MOVF001 & F004 ARE THE ONLY VALVES FROM GROUP 7 THAT ISOLATE ON THE STANDBY LIQUID CONTROL SYSTEM INITIATION SIGNAL

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

CONTAINMENT SYSTEMS

CONTAINMENT HUMIDITY CONTROL

LIMITING CONDITION FOR OPERATION

3.6.5.1 Containment temperature and relative humidity shall be maintained above the curve shown in Figure 3.6.5.1-1.

APPLICABILITY: Whenevery PRIMARY CONTAINMENT INTEGRITY is required for Specification 3.6.1.1.

ACTION:

1. With containment relative humidity not above the curve shown in Figure 3.6.5.1-1, close the containment purge inlet and outlet isolation valves within ___ hours.

2. With the containment temperature and relative humidity not above the curve shown in Figure 3.6.5.1-1:

   a. In OPERATIONAL CONDITION 1, 2 or 3, restore the temperature and relative humidity to a condition above the curve within ( ) hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. At all other times, either:

      1. Maintain an unobstructed opening(s) in the containment that equals or exceeds the flow area provided by two open vacuum breakers, or

      2. Deactivate the containment spray header isolation valves by ______________________________.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Containment temperature and relative humidity shall be verified to be above the curve shown in Figure 3.6.5.1-1:

a. At least once every 24 hours.

b. By verifying the temperature and humidity instrumentation 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.

(Initial) Temperature (°F)

FIGURE 3.6.5.1-1

CONTAINMENT TEMPERATURE
VS RELATIVE HUMIDITY

3/4.6.6 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.1 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

4.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the Annulus Building, the Auxiliary Building and the Fuel Building are less than or equal to 3.0, 0.25 and 0.25 inches of vacuum water gauge, respectively.

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatches and blowout panels are closed and sealed.

2. The door in each access to the secondary containment is closed, except for routine entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

*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)

c. At least once per 18 months:

1. Verifying that one standby gas treatment subsystem will draw down the Annulus Building and the Auxiliary Building to greater than or equal to 0.5 and 0.25 inches of vacuum water gauge in less than or equal to 169.5 and 45 seconds respectively, and,

2. Operating one standby gas treatment subsystem for one hour and maintaining the Annulus Building and the Auxiliary Building less than or equal to 3.0 and 0.25 inches of vacuum water gauge at a flow rate not exceeding 2000 and 5000 CFM, respectively.

3. Verifying that one fuel building variation subsystem will draw down the fuel building to less than -0.25 inches of vacuum water in less than or equal to 14.5 seconds, and

4. Operating one fuel building ventilation subsystem for one hour and maintaining less than or equal to -0.25 inches of vacuum water gauge in the fuel building at a flow rate not exceeding 5000 CFM.

SECONDARY CONTAINMENT AUTOMATIC ISOLATION (DAMPERS)(VALVES)

LIMITING CONDITION FOR OPERATION

3.6.6.2 The secondary containment ventilation system automatic isolation dampers valves shown in Table 3.6.6.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.6.2-1.

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.6.2-1 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 automatic damper secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

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

4.6.6.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.6.2-1 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.

*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. 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.

TABLE 3.6.6.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)	DAMPER GROUP*
1. Shield Building Annulus Ventilation Exhaust Damper (1 HVR and AOD 168)	15	12
2. Shield Building Annulus Ventilation Exhaust Damper (1 HVR and AOD)	15	12
3. Shield Building Annulus Ventilation Exhaust Damper (1 HVR and AOD 238)	15	12
4. Auxiliary Building Ventilation Exhaust Damper (1 HVR and AOD 214)	15	11
5. Auxiliary Building Ventilation Exhaust Damper (1 HVR and AOD 262)	15	11
6. Auxiliary Building Ventilation Exhaust Damper (1 HVR and AOD 249)	15	11
7. Auxiliary Building Ventilation Exhaust Damper (1 HVR and AOD 10A)	15	11
8. Auxiliary Building Ventilation Exhaust Damper (1 HVR and AOD 108)	15	11
9. Auxiliary Building Ventilation Supply Damper (1 HVR and AOD 143)	15	11
10. Auxiliary Building Ventilation Supply Damper (1 HVR and AOD 169)	15	11
11. Fuel Building Ventilation Supply Damper (1 HVR and AOD 132)	15	13
12. Fuel Building Ventilation Supply Damper (1 HVR and AOD 101)	15	13
13. Fuel Building Ventilation Supply Damper (1 HVR and AOD 132)	15	13
14. Fuel Building Ventilation Supply Damper (1 HVR and AOD 101)	15	13

TABLE 3.6.6.2-1
(Continued)

DAMPER FUNCTION	MAXIMUM ISOLATION TIME (Seconds)	DAMPER GROUP*
15. Fuel Building Ventilation Exhaust Damper (1 HVF and AOD 102)	15	13
16. Fuel Building Ventilation Exhaust Damper (1 HVF and AOD 112)	15	13

*See Table 3.3.2-1.

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.3 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

4.6.6.3 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.

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 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 12,500 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 12,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and

      b) Fuel handling accident.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches Water Gauge while operating the filter train at a flow rate of 12,500 cfm ± 10%.

   3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

SURVEILLANCE REQUIREMENTS (Continued)

a. Manual initiation from the control room, and

b. Simulated automatic imitation signal.

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5. Verifying that the heaters dissipate $\geq$ 61 kw when tested in accordance with ANSI N510-1975.

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 accordance with ANSI N510-1975 while operating the system at a flow rate of 12,500 cfm ± 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 N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 12,500 cfm ± 10%.

SHIELD BUILDING ANNULUS MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.4 Two independent Shield Building Annulus Mixing subsystems shall be OPERABLE.

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

ACTION:

a. With one Shield Building Annulus Mixing 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 Shield Building Annulus Mixing 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

4.6.6.4 Each Shield Building Annulus Mixing subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, verifying that the subsystem operates for at least (10 hours) and

*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 by:

1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

   a) LOCA, and

   b) Fuel handling accident.

2. Verifying that each subsystem was a flow rate of (52, 500C) CFM ± 10%.

3. Verifying that the subsystem starts and isolation dampers open on each of the following test signals:

   a. Manual initiation from the control room, and

   b. Simulated automatic imitation signal.

FUEL BUILDING VENTILATION

LIMITING CONDITION FOR OPERATION

3.6.6.5 Two independent Fuel Building Ventilation subsystems shall be OPERABLE.

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

ACTION:

a. With one Fuel Building Ventilation 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 Fuel Building Ventilation 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

4.6.6.5 Each Fuel Building Ventilation 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.

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 10,000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 10,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and

      b) Fuel handling accident.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 8 inches Water Gauge while operating the filter train at a flow rate of 10,000 cfm ± 10%.

   3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

SURVEILLANCE REQUIREMENTS (Continued)

a. Manual initiation from the control room, and

b. Simulated automatic imitation signal.

4. Verifying that the filter cooling bypass dampers can be manually opened and the fan can be manually started.

5. Verifying that the heaters dissipate 57 $\pm$ 5.7 kw when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter 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 10,000 cfm ± 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 N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 10,000 cfm ± 10%.

CONTAINMENT SYSTEMS

3/4.6.7 ATMOSPHERE CONTROL

PRIMARY CONTAINMENT HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.7.1 Two independent primary containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one primary containment hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.1 Each primary containment hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum (heater sheath) temperature increases to greater than or equal to (700)°F within (90) minutes (Upon reaching (700)°F, increase the power setting to maximum power for (2) minutes and verify that the power meter reads greater than or equal to (60) kW.) Maintain ≥ (700)°F for at least (2) hours.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all (control room) 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 during a recombiner system functional test that the (heater sheath) temperature increases to greater than or equal to (1200)°F within (5) hours and is maintained between ( )°F and ( )°F for at least 4 hours.)

   4. 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.)

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT/DRYWELL HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.2 Two independent primary containment/drywell hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one primary containment/drywell hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.2 Each primary containment/drywell hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   1. Starting the system from the control room, and

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least 513 cfm.

3/4.7 PLANT SYSTEMS

3/4.7.1 SERVICE WATER SYSTEMS

STANDBY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.1 At least two independent standby service water (SSW) system subsystems, with each subsystem comprised of:

a. Two OPERABLE SSW pump, and

b. An OPERABLE flow path capable of taking suction from the standby cooling tower basin and transferring the water through the RHR heat exchangers, ECCS pump room seal coolers, and associated coolers and pump heat exchangers,

shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, two subsystems.

b. In OPERATIONAL CONDITION 4, 5 and*, the subsystem(s) associated with systems and components required OPERABLE by Specifications 3.4.9.1, 3.4.9.2, (3.5.2), 3.9.11.1, 3.9.11.2 and 3.8.1.2.

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

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3:

1. With one SSW 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.

2. With both SSW subsystems inoperable, 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 SSW subsystem inoperable which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, as applicable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.

*When handling irradiated fuel in primary or secondary containment.

**Whenever both RHR shutdown cooling mode loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. In OPERATIONAL CONDITION 4 or 5 with the SSW subsystem inoperable which is associated with an ECCS pump required OPERABLE by Specification 3.5.2, declare the associated ECCS pump inoperable and take the ACTION required by Specification 3.5.2.

d. In OPERATIONAL CONDITION 5 with the SSW subsystem inoperable which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2.

e. In Operational Condition *, with the SSW subsystem inoperable which is associated with a diesel generator required OPERABLE by Specification 3.8.1.2, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system subsystem(s) shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that:

1. Each automatic valve servicing safety related equipment or isolating non-safety related equipment actuates to the correct position on a reactor plant cooling water system (RPCCW) low pressure signal and on a normal service water low pressure signal.

2. One pump in each subsystem starts on a:

    a) RPCCW low pressure signal and
    b) normal service water low pressure signal.

3. Each pump in each subsystem starts on a manual control signal from the main control room.

PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.2 The standby cooling water storage basin shall be OPERABLE with:

a. A minimum basin water level at or above elevation ( ) Mean Sea Level, USGS datum, and

b. An average basin water temperature of less than or equal to ( )°F.

(c. (At least) (Two) OPERABLE cooling tower fans.)

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

ACTION:

With the requirements of the above specification not satisfied:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one SSW (basin) inoperable, declare the associated SSW subsystem inoperable and, if applicable, declare the HPCS service water system inoperable, and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2, as applicable.

b. In OPERATIONAL CONDITION 4, 5 with both SSW (basins) inoperable, declare the SSW system and the HPCS service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In Operational Condition * with both SSW (basins) inoperable, declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.2 The standby cooling tower and water storage basin shall be determined OPERABLE at least once per:

a. 24 hours by verifying the basin water temperature and water level to be within their limits.

b. 31 days by starting each cooling tower fan from the control room and operating the fan for at least 15 minutes.

c. 18 months by verifying that each standby service water cooling tower fan starts automatically when the associated standby service water subsystem is initiated.

*When handling irradiated fuel in primary or secondary containment.

PLANT SYSTEMS

3/4.7.2 MAIN CONTROL ROOM AIR CONDITIONING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent main control room air conditioning system subsystems with air handling unit/filter train shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3 with one main control room air conditioning subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5 or *:

1. With one main control room air conditioning air handling/filter train subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the emergency mode of operation.

2. With both main control room air conditioning air handling/filter train subsystems inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary and 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 Each main control room air conditioning subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 104)°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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

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 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 4000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

e. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7 inches Water Gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that on each of the below emergency mode actuation test signals, the subsystem automatically switches to the emergency mode of operation and the control room is maintained at a positive pressure of $\geq$ 1/4 inch W.G. relative to the outside atmosphere during subsystem operation at a flow rate less than or equal to 4,000 cfm:

   a) LOCA, and

   b) Air intake radiation monitors.

3. Verifying that the heaters dissipate 23 ± 2.3 Kw when tested in accordance with ANSI N510-1975.

f. 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 accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

g. 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 N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

   3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 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 1020 + 40, - 20 psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

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

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 psig.*

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal and on a suppression pool water level - high signal.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

3/4.7.4 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.4 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 and OPERATIONAL CONDITIONS 4 and 5 for snubbers located on systems required OPERABLE in those OPERATIONAL CONDITIONS.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.4.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.4 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in lieu of the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection shall be performed at the first refueling outage. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications or damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing th next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.4.f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

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 or a visual inspection of the systems, within 72 hours for accessible areas and within 6 months for inaccessible areas following this determination. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

*The inspection interval for each type of snubbe 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.

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

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 plan for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing 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 4.7.4.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of type of that snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that 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 te 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) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

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.

SURVEILLANCE REQUIREMENTS (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.4.e for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers that fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers that have repairs that might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life 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 6.10.2.

FIGURE 4.7.4-1
SAMPLE PLAN FOR SNUBBER FUNCTIONAL TEST

PLANT SYSTEMS

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 10 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.5.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4.7.5.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

3/4.7.6 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 The fire suppression water system shall be OPERABLE with:

a. Three fire suppression pumps, each with a capacity of 1500 gpm, with their discharge aligned to the fire suppression header,

b. Two separate fire water tanks, each with a minimum contained volume of 265,000 gallons, and

c. An OPERABLE flow path capable of taking suction from both water storage tanks and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.7.2, 3.7.7.4, and 3.7.7.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 The fire suppression water system shall be demonstrated OPERABLE:

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

b. At least once per 31 days by starting the electric motor driven fire suppression pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that each fire suppression pump develops at least ______ gpm at a system head of _____ feet,

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that each fire suppression pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 70 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.6.1.2 Each diesel driven fire suppression pump shall be demonstrated OPERABLE:

a. At least once per 31 days by;

1. Verifying the fuel day tank contains at least 300 gallons of fuel.

2. Starting the pump from ambient conditions and operating 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.

4.7.6.1.3 Each diesel driven fire pump starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

    1. The electrolyte level of each cell is above the plates, and

    2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery. The specific gravity, corrected to 77°F and full electrolyte level, shall be greater than or equal to 1.20.

c. At least once per 18 months by verifying that:

    1. The batteries case and battery racks show no visual indication of physical damage or abnormal deterioration, and

    2. Battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.2 The following spray and sprinkler systems shall be OPERABLE:

LOCATION	ELEVATION	SYSTEM IDENTITY
a. Control Bldg. Cable Chases	116'0"	AS-6A
	98'0"	AS-6B
	70'0"	AS-6C, WS-6A, WS-6B, WS-6C
	115'0"	WS-7A, WS-7B
b. Cable Tunnels	67'6"/70'0"	WS-8D, WS-8E, WS-8F,
	67'6"/70'0"	WS-8G, WS-8H, WS-8K,
	67'6"/70'0"	WS-8L, WS-8M, WS-8N
c. Auxiliary Bldg., RCIC	70'0"	PS-1, WS-19
Pump Room	141'0"	WS-4A, WS-4B, WS-20, AS-12
d. Diesel Generator Bldg.	98'0"	PS-2A, PS-2B, PS-2C
e. Fuel Bldg., Railroad Bay	95'0"	AS-5
	148'0"	WS-5A, WS-5B

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 one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.6.2 Each of the above required spray and sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the automatic systems, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a simulated actuation test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity.

3. By a visual inspection of each deluge nozzle's spray area to verify that the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray and sprinkler header and verifying each open head spray and sprinkler nozzle is unobstructed.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.3 The following Halon system shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure:

a. Main control room Power Generation Control Complex (PGCC).

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

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.6.3 The above required Halon system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each 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 actuates, manually and automatically, upon receipt of a simulated actuation signal, and

2. Performance of a flow test through accessible headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.6.4 The fire hose stations shown in Table 3.7.7.4-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7.7.4-1 inoperable, provide gated wye(s) on the nearest OPERABLE hose stations(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.6.4 Each of the fire hose stations shown in Table 3.7.7.4-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the station.

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

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7.6.4-1

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK INDENTIFICATION
a. Reactor Building	114'0"	HR - 16, 22
	141'0"	HR - 17, 23
	162'3"	HR - 18, 19, 24, 25
	186'3"	HR - 20, 21, 26
b. Auxiliary Building	70'0" (Stairwell)	HR - 84
	95'9"	HR - 6, 7, 8, 9
	114'0"	HR - 10, 11
	141'0"	HR - 12, 13, 14, 15
	170'0"	HR - 80
c. Control Building	70'0"	HR - 85, 86, 87
	98'0"	HR - 88, 89, 90
	115'0" and 116'0"	HR - 91, 92, 93, 94
	135'0" (Stairwell)	HR - 96
d. Fuel Building	70'0"	HR - 1, 2, 82
	95'0"	HR - 3, 4
	113'0"	HR - 81
	148'0"	HR - 5
e. Pipe Tunnel	67'6"	HR - 83
f. Turbine Building	95'0"	HR - 50, 51
	123'6"	HR - 53

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.6.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7.6.5-1 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7.7.5-1 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise provide the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.6.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7.7.5-1 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months, 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 than 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.

TABLE 3.7.6.5-1

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION		HYDRANT NUMBER
a.	Northeast of Fuel Bldg	FHY 11
b.	Northeast of Diesel Generator Bldg	FHY 27
c.	East of Control Bldg	FHY 13
d.	West of Standby Cooling Tower	FHY 9
e.	North of Fuel Bldg	FHY 10

3/4.7.7 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.7 All fire barrier 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 one 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) or 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

4.7.7.1 Each of the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE at least once per 18 months by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly.

b. Each fire damper and associated hardware.

c. At least 10 percent of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of each type of sealed penetration shall be made. This inspection process shall continue until a 10 percent sample 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.

4.7.7.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. That each locked-closed fire door is closed at least once per 7 days.

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and by performing a functional test of these mechanisms at least once per 18 months.

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

3/4.7.8 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.8 The temperature of each area shown in Table 3.7.9-1 shall be maintained within the limits indicated in Table 3.7.9-1.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7.9-1:

a. For more than eight hours, in lieu of any report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.8 The temperature in each of the areas shown in Table 3.7.9-1 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7.8-1

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
a.	
b.	
c.	
d.	
e.	

PLANT SYSTEMS

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

ACTION:

With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.9 The main turbine bypass system shall be demonstrated OPERABLE at least once per:

a. 7 days 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 that TURBINE BYPASS SYSTEM RESPONSE TIME meet the following requirements when measured from the initial movement of the main turbine stop valve and the main turbine control valve:

a) Main turbine bypass valve opening shall start within 0.1 seconds, and

b) at least 80% of main turbine bypass capacity shall be established within 0.3 seconds.

3/4.7.10 STRUCTURAL SETTLEMENT

LIMITING CONDITION FOR OPERATION

3.7.10 Structural settlement of the following structures shall be within the predicted values as shown in Figure 3.7.10-1.

a. Reactor Building
b. Auxiliary Building
c. Fuel Building
d. Control Building
e. Diesel Generator Building
f. Standby Cooling Tower, Basin and Pump House

APPLICABILITY: At all times.

ACTION:

With the measured structual settlement of any of the above required structures outside of the predicted settlement, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the settlement measurements and the predicted settlement, an analysis to demonstrate the continued structural integrity of the affected structure(s) and plans to monitor the settlement of the affected structure(s) in the future.

SURVEILLANCE REQUIREMENTS

4.7.10 The structural settlement of the above required structures shall be demonstrated to be within the predicted settlement values:

a. At least once per 92 days, using at least three markers per structure, until there is essentially no movement during those 92 days.

b. At least once per 24 months, using at least one marker per structure, for at least 10 years.

c. Following any seismic event equal to or greater than an Operational Basis Earthquake (OBE), using at least three markers per structure.

FIGURE 3.7.10-1

PREDICTED STRUCTURAL SETTLEMENT

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

   1. A separate day fuel tank containing a minimum of 316.3 gallons of fuel, equivalent to a level of (   )%,

   2. A separate fuel storage system containing a minimum of 45,495 gallons of fuel, equivalent to a level of (   )%, and

   3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator 1A or 1B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and diesel generators 1A and 1B to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one offsite circuit and diesel generator 1A or 1B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators 1A and 1B to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION (Continued)

c. With diesel generator 1C of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a 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 diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

d. With diesel generator 1A, 1B or 1C of the above required A.C. electrical power sources inoperable, in addition to ACTION a, b or c, as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With two of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4, for one diesel generator at a time, within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours. With only one offsite circuit restored to OPERABLE status, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

f. With diesel generators 1A and 1B of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators 1A and 1B to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore both diesel generators 1A and 1B to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each 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 and fuel tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 450 rpm for diesel generators 1A and 1B and 900 rpm for diesel generator 1C in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± -3 Hz for diesel generator 1A and 1B and 4160 ± 420 volts and 60 ± 1.2 Hz for diesel generator 1C 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 3500 kw for diesel generators 1A and 1B and 2600 kw for diesel generator 1C in less than or equal to 60 seconds, and operates with this load for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 250 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 fuel tanks.

c. At least once per 31 days by removing accumulated water from the fuel storage tank(s).

d. At least once per 31 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 from new fuel or prior to addition to the storage tank, as applicable, verify in accordance with the tests specified in 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 _____ but less than or equal to ____ or an API gravity @ 60°F of greater than or equal to _____ degrees but less than or equal to ____ 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 917.5 kw for diesel generator 1A, greater than or equal to 509.2 kw for diesel generator 1B, and greater than or equal to 1995 kw for diesel generator 1C while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 3 Hz for diesel generators 1A and 1B, and voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz for diesel generator 1C, and

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 3500 kw for diesel generators 1A and 1B and 2600 kw for diesel generator 1C without tripping. The generator voltage shall not exceed 4784 volts for diesel generators 1A and 1B or 58[illegible]4 volts for diesel generator 1C during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

   a) For divisions I and II:

      1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the 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 ± 3 Hz during this test.

   b) For division III:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with the permanently connected loads within 10 seconds 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 bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 3 Hz for diesel generators 1A and 1B, and 4160 ± 420 volts and 60 ± 1.2 Hz for diesel generator 1C 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. Verifying that on a simulated loss of the diesel generator, with offsite power not available, the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) For divisions I and II:

      1) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the 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 ± 3 Hz during this test.

   b) For division III:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts on the auto-start signal, energizes the emergency bus with its loads and the auto-connected emergency loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

8. Verifying that all automatic diesel generator trips are automatically bypassed upon loss of voltage on the emergency bus concurrent with an ECCS actuation signal except:

   a) For divisions 1 and 2, (engine overspeed and generator differential current).

   b) For division 3, (engine overspeed and generator differential current).

9. 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 3850 kw for diesel generators 1A and 1B and 2850 kw for diesel generator 1C. During the remaining 22 hours of this test, the diesel generator shall be loaded to 3500 kw for diesel generator 1A and 1B and 2600 kw for diesel generator 1C. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 3 Hz for diesel generators 1A and 1B and 4160 ± 420 volts and 60 ± 1.2 Hz for diesel generator 1C 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.a)2) and b)2)*.

10. Verifying that the auto-connected loads to each diesel generator do not exceed the 8760-hour rating of 3500 kw for diesel generator 1A and 1B and the 2000-hour rating of 2850 kw for diesel generator 1C.

11. 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.

12. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.

13. Verifying that with all diesel generator air start receivers pressurized to less than or equal to 250 psig and the compressors secured, the diesel generator starts at least 5 times from ambient conditions and accelerates to 900 rpm ± 3% in less than or equal to 10 seconds.

*If Surveillance Requirements 4.8.1.1.2.e(4).a)2 and b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at 3500 kw for diesel generators 1A and 1B and 2600 kw for diesel generator 1C for one hour or until operating temperatures have stabilized.

14. Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 1A and 1B.

15. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) (Turning gear engaged.)

    b) (Emergency stop.)

f. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that all three diesel generators accelerate to at least 450 rpm for diesel generators 1A and 1B and 900 rpm for diesel generator 1C in less than or equal to 10 seconds.

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 pursuant to Specification 6.9.1. 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. Diesel generator 1A or 1B, and diesel generator 1C when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. A day fuel tanks containing a minimum of 316.3 gallons of fuel, equivalent to a level of (   )%.

   2. A fuel storage system containing a minimum of 44,376 gallons of fuel, equivalent to a level of (   )%.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the offsite circuits and/or diesel generators 1A or 1B of the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 23 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator 1C of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator 1C to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3

c. The provisions of Specification 3.0.3 are not applicable.

*When handling irradiated fuel in the primary or secondary containment.

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.

ELECTRICAL POWER SYSTEMS

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. 125 volt battery 1A.
   2. 125 volt full capacity Class 1E source charger.

b. Division II, consisting of:
   1. 125 volt battery 1B.
   2. 125 volt full capacity Class 1E source charger.

c. Division III, consisting of:
   1. 125 volt battery 1C.
   2. 125 volt full capacity Class 1E source charger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With either Division I or Division II battery and/or charger of the above required D.C. electrical power sources inoperable, restore the inoperable division to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With Division III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125-volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and

   2. Total battery terminal voltage is greater than or equal to 130-volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

   1. The parameters in Table 4.8.2.1-1 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 x 10^{-6} ohms, and

   3. The average electrolyte temperature of a representative number of connected cells is above 60°F.

c. At least once per 18 months by verifying that:

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

   2. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material,

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10^{-6} ohms, and

   4. The battery charger will supply at least 300 amperes for chargers 1A and 1B and 50 amperes for charger 1C at a minimum of 130 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 in accordance with IEEE 450 while maintaining the battery terminal voltage greater than or equal to 105 volts.

      a) Division 1
         $\geq$ 671 amperes for the first 60 seconds
         $\geq$ 270 amperes for the next 9 minutes
         $\geq$ 336 amperes for the next 60 seconds
         $\geq$ 270 amperes for the next 228 minutes
         $\geq$ 451 amperes for the last 60 seconds

b) Division 2
≥ 502 amperes for the first 60 seconds
≥ 261 amperes for the next 9 minutes
≥ 327 amperes for the next 60 seconds
≥ 261 amperes for the next 228 minutes
≥ 327 amperes for the last 60 seconds

c) Division 3
≥ 77.4 amperes for the first 60 seconds
≥ 11.4 amperes for the next 234 minutes

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. 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.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	$\geq$ 1.200(b)	$\geq$ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amperes when on float charge.
(c) May be corrected for average electrolyte temperature.
(1) For 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 7 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.

D.C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, Division I or Division II, and, when the HPCS system is required to be OPERABLE, Division III, of the D.C. electrical power sources shall be OPERABLE with:

a. Division I consisting of:
   1. 125 volt battery 1A.
   2. 125 volt full capacity Class 1E source charger.

b. Division II consisting of:
   1. 125 volt battery 1B.
   2. 125 volt full capacity Class 1E source charger.

c. Division III consisting of:
   1. 125 volt battery 1C.
   2. 125 volt full capacity Class 1E source charger.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the Division I and/or Division II battery and/or charger of the above required D.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment and operations with a potential for draining the reactor vessel.

b. With Division III battery and/or charger of the above required D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required battery and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

*When handling irradiated fuel in the primary or secondary containment.

ELECTRICAL POWER SYSTEMS

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 (both) between redundant buses within the unit (and between units at the same station):

a. A.C. power distribution:

1. Division I, consisting of:

   a) 4160 volt A.C. bus IENS*SWG1A
   b) 480 volt A.C. switchgear IEJS*SWG1A and 1EJS*SWG2A.
   c) 480 volt A.C. MCCs 1EHS*MCC 2A, 2C, 2E, 2G, 2J, 2L, 8A, 14A, 15A and 16A.
   d) 120 volt A.C. distribution panels 1SCV*PNL2A1, 2A2, 2C1, 2E1, 2G1, 2J1, 2L1, 8A1, 14A1, 15A1, and 16A1 in (480 volt MCCs ____, ____, ____ and ____) with 1SCM*PNL01A energized from voltage regulating transformer 1SCM*XRC14A1, and 1VBS*PNL01A energized from uninterruptible power supply inverter 1ENB*INV01A connected to D.C. Division I# 125 volt D.C. bus 1ENB*SWG01A and 480 volt A.C. MCCs 1EHS*MCC8A and 14A.

2. Division II consisting of:

   a) 4160 volt A.C. bus 1ENS*SWG1B.
   b) 480 volt A.C. switchgear 1EJS*SWG1B and 1EJS*SWG2A.
   c) 480 volt A.C. MCCs 1 EHS*MCC 2B, 2D, 2F, 2H, 2K, 8B, 14B, 15B and 16B.
   d) 120 volt A.C. distribution panels 1SCV*PNL2B1, 2B2, 2D1, 2F1, 2H1, 2K1, 8B1, 14B1, 15B1 and 16B1 in (480 volt MCCs ____, ____, ____ and ____) with 1 SCM*PNL01B energized from voltage regulating transformer 1SCM*XRC14B1, and 1VBS*PNL01B energized from uninterruptible power supply inverter 1ENB*INV01B connected to D.C. Division II# 125 volt D.C. Bus 1ENB*SWG01B and 480 volt A.C. MCCs 1EHS*MCC 8B and 14B.

3. Division III, consisting of:

   a) 4160 volt A.C. bus IG22*S004
   b) 480 volt A.C. switchgear 1E22*S002
   c) 120 volt A.C. distribution panel 1E22*S002PNL in 480 volt MCC 1E22*S002

LIMITING CONDITION FOR OPERATION (Continued)

b. D.C. power distribution:

1. Division I, consisting of 125 volt D.C. Bus 1ENB*SWG01A, distribution panels 1ENB*PNL02A, and 1ENB*PNL03A and MCC 1ENB*MCC1.

2. Division II, consisting of 125 volt D.C. BUS 1 ENB*SWG01B and distribution panels 1ENB*PNL02B and 1ENB*PNL03B.

3. Division III, consisting of 125 volt D.C. distribution panel 1E22*S001PNL.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. For A.C. power distribution:

1. With either Division I or Division II of the above required A.C. distribution system not energized, re-energize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division III of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

3. With one of the above required inverters inoperable, energize the associated distribution panel within 8 hours; restore the inoperable inverter to OPERABLE and energized status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For D.C. power distribution:

1. With either Division I or Division II of the above required D.C. distribution system not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division III of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

*One inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on the associated battery bank provided (1) its buses/MCCs/panels are OPERABLE and energized, and (2) the buses/MCCs/panels associated with the other battery banks are OPERABLE and energized.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment on the busses/switchgear/MCCs/panels and voltage on the busses/ switchgear/ MCCs/panels.

ELECTRICAL POWER SYSTEMS

DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following power distribution system divisions shall be energized:

a. For A.C. power distribution, Division I or Division II, and when the HPCS system is required to be OPERABLE, Division III, with:

   1. Division I consisting of:

      a) 4160 volt A.C. bus 1ENS*SWG1A.
      b) 480 volt A.C. switchgear 1EJS*SWG1A and 1EJS*SWG2A.
      c) 480 volt A.C. MCCs 1EHS*MCC 2A, 2C, 2E, 2G, 2J, 2L, 8a, 14A, 15A and 16A.
      d) 120 volt A.C. distribution panels 1SCV*PNL2A1, 2A2, 2C1, 2E1, 2G1, 2J1, 2L1, 8A1, 14A1, 15A1, and 16A1 in (480 volt MCCs ____, ____, ____ and ____) with 1SCM*PNL01A energized from voltage regulating transformer 1SCM*XRC14A1, and 1VBS*PNL01A energized from uninterruptible power supply inverter 1ENB*INV01A connected to D.C. Division I# 125 volt D.C. Bus 1ENB*SWG01A and 48 volt A.C. MCCs 1EHS*MCC 8A and 14A.

   2. Division II consisting of:

      a) 4160 volt A.C. bus 1ENS*SWG1B.
      b) 480 volt A.C. switchgear 1EJS*SWG1B and 1EJS*SWG2B.
      c) 480 volt A.C. MCCs 1EHS*MCC 2B, 2D, 2F, 2H, 2K, 8B, 14B, 15B and 16B.
      d) 120 volt A.C. distribution panels 1SCV*PNL2B1, 2B2, 2D1, 2F1, 2H1, 2K1, 8B1, 14B1, 15B1 and 16B1 in (480 volt MCCs ___, ___, ___ and ___) with 1SCM*PNL01B energized from voltage regulating transformer 1SCM*XRC14B1, and 1VBS*PNL01B energized from uninterruptible power supply inverter 1ENB*INV01B connected to D.C. Division II# 125 volt D.C. Bus 1ENB*SWG01B and 480 volt A.C. MCCS 1EHS*MCC 8B and 14B.

   3. Division III consisting of:

      a) 4160 volt A.C. bus 1E22*S004.
      b) 480 volt A.C. switchgear 1E22*S002.
      c) 120 volt A.C. distribution panel 1E22*S002PNL and 480 volt MCC 1E22*S002.

b. For D.C. power distribution, Division I or Division II, and when the HPCS system is required to be OPERABLE, Division III, with:

   1. Division I consisting of 125 volt D.C. bus 1ENB*SWG01A and distribution panel 1ENB*PNL02A and 1ENB*PNL03A and MCC 1 ENB*MCC1.

*When handling irradiated fuel in the primary or secondary containment.

LIMITING CONDITION FOR OPERATION (Continued)

2. Division II consisting of 125 volt D.C. bus 1ENB*SWG01B and distribution panels 1ENB*PNL02B and 1ENB*PNL03B.

3. Division III consisting of 125 volt D.C. distribution panel 1E22*S001PNL.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. For A.C. power distribution:

   1. With less than Division I and/or Division II of the above required A.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment and operations with a potential for draining the reactor vessel.

   2. With Division III of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

b. For D.C. power distribution:

   1. With less than Division I and/or Division II of the above required D.C. distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary or secondary containment and operations with a potential for draining the reactor vessel.

   2. With Division III of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment on the busses/switchgear/MCCs/panels and voltage on the busses/switchgear/ MCCs/panels.

*When handling irradiated fuel in the primary or secondary containment.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 4.16 kV circuit breakers, de-energize the 4.16 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 480 volt circuit breakers, remove the inoperable circuit breaker(s) from service by racking out the breaker within 72 hours and verify the inoperable breaker(s) to be racked out at least once per 7 days thereafter.

   3. For 480 volt MCC circuit breaker/fuse combination starters, remove the inoperable starter(s) from service by locking the breakers open and removing the control power fuse within 72 hours and verify the inoperable starter(s) circuit breaker to be locked open with the control power fuse 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 redundant circuit breakers tripped or to 480 volt circuits which have the inoperable circuit breaker racked out.

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices shown in Table 3.8.4.1-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage 4.16 KV circuit breakers are OPERABLE by selecting, on a rotating basis, at least one of the four circuit breakers and performing:

    a) A CHANNEL CALIBRATION of the associated protective relays, and

    b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed.

    c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least one of the four circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the long time delay trip element and 150% of the pickup of the short time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ± 20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

SURVEILLANCE REQUIREMENTS (Continued)

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional testing shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION — SYSTEM(S) AFFECTED

a. (4.16) KV Circuit Breakers

	PRIMARY PROTECTION		SECONDARY PROTECTION		EQUIPMENT ID#
	LOCATION	DEVICE #	LOCATION	DEVICE #	
1.	IENS SWG 4A	ACB 3B	IENS SWG 3A	ACB 35	1033-C001A
2.	IENS SWG 4B	ACB 3B	IENS SWG 3B	ACB 37	1033-C001A

b. (480)VAC (Molded Case) Circuit Breakers

1. Type Square D

Location	Location	Equip. No.
ILAR-BKR 1B	ILAR-BKR 1A	ILAR-PNL1R1
ILAR-BKR 2B	ILAR-BKR 2A	ILAR-PNL1R:
ILAR-BKR 3B	ILAR-BKR 3A	ILAR-PNL1R3
ILAR-BKR 4B	ILAR-BKR 4A	ILAR-PNL1R4
ILAR-BKR 5B	ILAR-BKR 5A	ILAR-PNL1R5
ILAR-BKR 6B	ILAR-BKR 6A	ILAR-PNL1R6
ILAR-BKR 7B	ILAR-BKR 7A	ILAR-PNL1R7
ILAR-BKR 8B	ILAR-BKR 8A	ILAR-PNL1R8
ILAR-BKR 9B	ILAR-BKR 9A	ILAR-PNL1R9
ILAR-BKR 10B	ILAR-BKR 10A	ILAR-PNL1R10
ILAR-BKR 11B	ILAR-BKR 11A	ILAR-PNL1R11
ILAR-BKR 12B	ILAR-BKR 12A	ILAR-PN'1R12

TABLE 8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION		SYSTEM(S) AFFECTED
PRIMARY PROTECTION	SECONDARY PROTECTION	EQUIPMENT ID#
LOCATION / DEVICE #	LOCATION / DEVICE #	
Location	Location	Equip. No.
b. (480)VAC (Molded Case) Circuit Breakers (continued)		
ILAR-BKR 13B	ILAR-BKR 13A	ILAR-PNL1R13
ILAR-BKR 14B	ILAR-BKR 14A	ILAR-PNL1R14
ILAR-BKR 15B	ILAR-BKR 15A	ILAR-PNL1R15
ILAR-BKR 16B	ILAR-BKR 16A	ILAR-PNL1R16
ILAR-BKR 17B	ILAR-BKR 17A	ILAR-PNL1R17
ILAR-BKR 18B	ILAR-BKR 18A	ILAR-PNL1R18
ILAR-BKR 19B	ILAR-BKR 19A	ILAR-PNL1R19
ISCA-BKR 2A12	ISCA-BKR 2A11	ISCA-PNL2A1
ISCA-BKR 2D12	ISCA-BKR 2D11	ISCA-PNL2D1
ISCA-BKR 8A22	ISCA-BKR 8A21	ISCA-PNL8A2
ISCA-BKR 8B22	ISCA-BKR 8B21	ISCA-PNL8B2

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
c. Gould Circuit Breaker Type AB21		Gould Starter/Controller Type FVNR Size 1		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
IEHS*MCC 2A	2B	IEHS MCC 2A	2B	ICAM*FN1A
IEHS*MCC 2B	2B	IEHS MCC 2B	2B	ICAM*FN1B
INHS-MCC 2A	2A	INHS-MCC 2A	2A	IC41-D002
INHS-MCC 2A	3C	INHS-MCC 2A	3C	IDER-P1A
INHS-MCC 2A	3D	INHS-MCC 2A	3D	IDER-P2A
INHS-MCC 2A	4D	INHS-MCC 2A	4D	IDFR-P2A
INHS-MCC 2A	4E	INHS-MCC 2A	4E	IDFR-P1A
INHS-MCC 2B	4C	INHS-MCC 2B	4C	IDER-P1B
INHS-MCC 2B	5C	INHS-MCC 2B	5C	IDER-P2B
INHS-MCC 2B	6B	INHS-MCC 2B	6B	IDFR-P2B
INHS-MCC 2A	6C	INHS-MCC 2B	6C	IHVR-FNID
INHS-MCC 2C	1E	INHS-MCC 2C	1E	1B33-COOLBH
INHS-MCC 2D	3B	INHS-MCC 3D	3B	1B33-COOLBH
INHS-MCC 2E	2B	INHS-MCC 2E	2B	1G36-C002
INHS-MCC 2E	2C	INHS-MCC 2E	2C	INVE-FNIC
INHS-MCC 2E	3B	INHS-MCC 2E	3B	1G36-C001A
INHS-MCC 2E	4D	INHS-MCC 2E	4D	INCS-P5A
INHS-MCC 2E	4E	INHS-MCC 2E	4E	1B33-D003A2
INHS-MCC 2E	6C	INHS-MCC 2E	6C	1B33-D003A5
INHS-MCC 2E	1C	INHS-MCC 2E	1C	1G36-
INHS-MCC 2F	3B	INHS-MCC 2F	3B	1G33-C001B
INHS-MCC 2F	3C	INHS-MCC 2F	3C	IHVR-FNIB

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
Gould Circuit Breaker Type AB21		Gould Starter/Controller Type FVNR Size 1		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
c. (continued)				
INHS-MCC 2F	4A	INHS-MCC 2F	4A	IDFR-P1B
INHS-MCC 2F	5A	INHS-MCC 2F	5A	IWCS-P5B
INHS-MCC2F	5C	INHS-MCC2F	5C	1B33-D003B5
INHS-MCC 2F	6B	INHS-MCC 2F	6B	1B33-D003B2
INHS-MCC2F	6C	INHS-MCC 2F	6C	IG36-A002AG
INHS-MCC 8A	2E	INHS-MCC 8A	2E	1F42-D002
INHS-MCC 8A	3E	INHS-MCC 8A	3E	IDFR-P6A
INHS-MCC 8B	3C	INHS-MCC 8B	3C	IDFR-P6B
INHS-MCC 102B	3A	INHS-MCC 102B	3A	ICPP-FN1
INHS-MCC 7A	6F	INHS-MCC 7A	6F	IHVR-FN1A

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
d. Gould Circuit Breaker Type AB22		Gould Starter/Controller Type FVR Size 1		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
1EHS*MCC 2A	2A	IEHS*MCC 2A	2A	IC41*F001A
1EHS*MCC 2A	5A	IEHS*MCC 2A	5A	ISWP*MCV4A
1EHS*MCC 2A	5B	IEHS*MCC 2A	5B	ISWP*MCV5B
1EHS*MCC 2A	5C	IEHS*MCC 2A	5C	ISWP*MCV502A
1EHS*MCC 2A	6A	IEHS*MCC 2A	6A	IRCS*MOV58A
1EHS*MCC 2A	6B	IEHS*MCC 2A	6B	IRCS*MOV59
1EHS*MCC 2A	6C	IEHS*MCC 2A	6C	ISLOP*MOV503A
1EHS*MCC 2B	1B	IEHS*MCC 2B	1B	ISFC*MOV120
1EHS*MCC 2B	1D	IEHS*MCC 2B	1D	ISFC*MOV139
1EHS*MCC 2B	2A	IEHS*MCC 2B	2A	IC41*F001B
1EHS*MCC 2B	5A	IEHS*MCC 2B	5A	ISWP*MOV4B
1EHS*MCC 2B	5B	IEHS*MCC 2B	5B	ISWP*MOV5A
1EHS*MCC 2B	5C	IEHS*MCC 2B	5C	ISWP*MOV502B
1EHS*MCC 2B	6A	IEHS*MCC 2B	6A	IRCS*MOV58B
1EHS*MCC 2B	6B	IEHS*MCC 2B	6B	IRCS*MOV59B
1EHS*MCC 2B	6C	IEHS*MCC 2B	6C	ISWP*MOV59B
1EHS*MCC 2C	1D	IEHS*MCC 2C	1D	ICCP*MOV142
1EHS*MCC 2C	2C	IEHS*MCC 2C	2C	ICCP*MOV143
1EHS*MCC 2C	2D	IEHS*MCC 2C	2D	ICPM*MOV1A
1EHS*MCC 2C	3A	IEHS*MCC 2C	3A	ICPM*MOV2A
1EHS*MCC 2C	3B	IEHS*MCC 2C	3B	ICPM*MOV3A
1EHS*MCC 2C	3C	IEHS*MCC 2C	3C	1E12*F037A

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
Gould Circuit Breaker Type AB22		Gould Starter/Controller Type FVNR Size 1		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
d. (continued)				
1EHS*MCC 2C	4A	IEHS*MCC 2C	4A	1E12*F042A
1EHS*MCC 2C	4B	IEHS*MCC 2C	4B	IRVN*MOV22A
1EHS*MCC 2C	4C	IEHS*MCC 2C	4C	IRCS*MOV60A
1EHS*MCC 2C	5B	IEHS*MCC 2C	5B	IRCS*MOV61A
1EHS*MCC 2C	5C	IEHS*MCC 2C	5C	ICPM*MOV4A
1EHS*MCC 2D	1C	IEHS*MCC 2D	1C	____*F016
1EHS*MCC 2D	1D	IEHS*MCC 2D	1D	ICPM*MOV1B
1EHS*MCC 2D	2C	IEHS*MCC 2D	2C	ICPM*MOV2B
1EHS*MCC 2D	2D	IEHS*MCC 2D	2D	ICPM*MOV3B
1EHS*MCC 2D	3A	IEHS*MCC 2D	3A	ICPM*MOV4B
1EHS*MCC 2D	3B	IEHS*MCC 2D	3B	ICPP*MOV104
1EHS*MCC 2D	3C	IEHS*MCC 2D	3C	1E51*F063
1EHS*MCC 2D	4A	IEHS*MCC 2D	4A	1E51*F076
1EHS*MCC 2D	4B	IEHS*MCC 2D	4B	1G63*F001
1EHS*MCC 2D	4C	IEHS*MCC 2D	4C	1G63*F028
1EHS*MCC 2D	5A	IEHS*MCC 2D	5A	IWCS*MOV178
1EHS*MCC 2K	1D	IEHS*MCC 2K	1D	ICCP*MOV144
1EHS*MCC 2K	2A	IEHS*MCC 2K	2A	IRCS*MOV60B
1EHS*MCC 2K	2B	IEHS*MCC 2K	2B	IRCS*MOV611
1EHS*MCC 2K	2C	IEHS*MCC 2K	2C	IHVN*MOV22B

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
Gould Circuit Breaker Type AB22		Gould Starter/Controller Type FVR Size 1		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
d. (continued)				
1EHS*MCC 2K	3D	IEHS*MCC 2K	3D	1E12*F042B
1EHS*MCC 2K	4A	IEHS*MCC 2K	4A	1E12*F009
1EHS*MCC 2K	4D	IEHS*MCC 2K	4D	1G33-F053
1EHS*MCC 2K	5A	IEHS*MCC 2K	5A	1G33-F040
1EHS*MCC 2K	6C	IEHS*MCC 2K	6C	IHVN*MOV102
1EHS*MCC 2K	6D	IEHS*MCC 2K	6D	1E12*F037B
1EHS*MCC 2K	7D	IEHS*MCC 2K	7D	ICCP*MOV158
INHS*MCC 2A	1C	INHS*MCC 2A	1C	IB21-F001
INHS*MCC 2A	1D	INHS*MCC 2A	1D	IB33-F02A
INHS*MCC 2A	5C	INHS*MCC 2A	5[illegible]	1G33-F102
INHS*MCC 2A	5D	INHS*MCC 2A	[illegible]	1G33-F067A
INHS*MCC 2A	7D	INHS*MCC 2A	7D	1G33-F106
INHS*MCC 2B	3B	INHS*MCC 2B	3B	1G33-F042
INHS*MCC 2B	3C	INHS*MCC 2B	3C	1B21-F002
INHS*MCC 2B	4D	INHS*MCC 2B	4D	1G33-F044
INHS*MCC 2B	5D	INHS*MCC 2B	5D	1G33-F100
INHS*MCC 2B	6D	INHS*MCC 2B	6D	1G33-F101
INHS*MCC 2D	2E	INHS*MCC 2D	2E	1B21-F005
INHS*MCC 2D	3D	INHS*MCC 2D	3D	1B33-F067B
INHS*MCC 2D	4D	INHS*MCC 2D	4D	1B33-F023B
INHS*MCC 2E	3A	INHS*MCC 2E	3A	1G33-F031
INHS*MCC 2E	5E	INHS*MCC 2E	5E	1G33-F107
INHS*MCC 2F	2D	INHS*MCC 2F	2D	1G33-F104
INHS*MCC 8A	4E	INHS*MCC 8A	4E	1C11-F003

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
e. Gould Circuit Breaker Type HE43		Gould Starter/Controller Type Later		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
INHS*MCC 2A	2C	INHS*MCC 2A	2C	1POP-WR2A01
INHS*MCC 2A	2D	INHS*MCC 2A	2D	1POP-WR2A02
INHS*MCC 2D	5C	INHS*MCC 2D	5C	1POP-WR2D01
INHS*MCC 2D	5D	INHS*MCC 2D	5D	1POP-WR2D02
INHS*MCC 8A	1E	INHS*MCC 8A	1E	1F15-E003
INHS*MCC 8A	2D	INHS*MCC 8A	2D	1F15-E005
INHS*MCC 8A	4C	INHS*MCC 8A	4C	1F11-E012
INHS*MCC 8A	6B	INHS*MCC 8A	6B	1FNR-P06
INHS*MCC 8A	6C	INHS*MCC 8A	6C	1FNR-P08
INHS*MCC 8B	2A	INHS*MCC 8A	2A	1FNR-P07
INHS*MCC 2F	2A	INHS*MCC 2F	2A	1POP-WR2F01
INHS*MCC 2F	2B	INHS*MCC 2F	2B	1SRB-EL1A

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION | SYSTEM(S) AFFECTED

f. PRIMARY PROTECTION: Gould Circuit Breaker Type A80 — SECONDARY PROTECTION: Gould Starter/Controller Type FVNR Size 3

Location (Primary)	Cubicle (Primary)	Location (Secondary)	Cubicle (Secondary)	EQUIP. NO.
IEHS*MCC 2A	2C	IEHS*MCC 2A	2C	1C41*C001A
IEHS*MCC 2B	2C	IEHS*MCC 2B	2C	1C41*C001B
INHS*MCC 2B	2D	INHS*MCC 2B	2D	1C41*D003
INHS*MCC 2E	1D	INHS*MCC 2E	1D	1B33-D003A1
INHS*MCC 2E	6D	INHS*MCC 2E	6D	1B33-D003A4
INHS*MCC 2F	4D	INHS*MCC 2F	4D	1B33-D003B1
INHS*MCC 2F	6D	INHS*MCC 2F	6D	1B33-D003B4

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION

SYSTEM(S) AFFECTED

PRIMARY PROTECTION — SECONDARY PROTECTION

g. Gould Circuit Breaker Type ABO — Gould Starter/Controller Type 25P1W Size 4 — EQUIP. NO.

Location	Cubicle	Location	Cubicle	EQUIP. NO.
INHS*MCC 102A	1C	INHS*MCC 102A	1C	1DRS-UC1A
INHS*MCC 102A	2C	INHS*MCC 102A	2C	1DRS-UC1C
INHS*MCC 102A	3B	INHS*MCC 102A	3B	1DRS-UC1E
INHS*MCC 102B	1C	INHS*MCC 102B	1C	1DRS-UC1B
INHS*MCC 102B	2C	INHS*MCC 102B	2C	1DRS-UC1D
INHS*MCC 102B	3B	INHS*MCC 102B	3B	1DRS-UC1F

PRIMARY PROTECTION — SECONDARY PROTECTION

h. Gould Circuit Breaker Type A821 — Gould Starter/Controller Type FVNR Size 2 — EQUIP. NO.

Location	Cubicle	Location	Cubicle	EQUIP. NO.
INHS*MCC 8B	1D	INHS*MCC 8B	1D	1F42-CRNE001

TABLE 8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTION DEVICES

DEVICE NUMBER AND LOCATION				SYSTEM(S) AFFECTED
PRIMARY PROTECTION		SECONDARY PROTECTION		
i. Gould Circuit Breaker Type (Later)		Gould Circuit Breaker Type (Later)		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
IEJS*LDC 2B	(Later)	1EJS*LDC 2B	(Later)	1HVR-UC1C
IEJS*LDC 2A	(Later)	1EJS*LDC 2A	(Later)	1HVR-UC1A
IEJS*LDC 2A	(Later)	1EJS*LDC 2A	(Later)	1MNR-SWI
IEJS*LDC 2B	(Later)	1EJS*LDC 2B	(Later)	1HVR-UC1B
1H22-P008	(Later)		(Later)	1C51-S0016
1H22-P008	(Later)		(Later)	1C51-S0014
1H22-P008	(Later)		(Later)	1C51-S001B
1H22-P008	(Later)		(Later)	1C51-S001D
1H22-P008	(Later)		(Later)	1C51-S001S
1H22-P008	(Later)		(Later)	1C51-S001K
1H22-P008	(Later)		(Later)	1C51-S001L
1H22-P008	(Later)		(Later)	1C51-S001M
1H13*P703	(Later)		(Later)	IH22*PNL071
IMCS*PWRS1A	(Later)		(Later)	IMCS*RBNR1A
PRIMARY PROTECTION		SECONDARY PROTECTION		
i. GE Circuitry Type (Later)		GE Circuitry Type (Later)		EQUIP. NO.
Location	Cubicle	Location	Cubicle	
1HCS*PWRS1B	(Later)		(Later)	1HCS*RBNR1B
1H13*P702	(Later)		(Later)	1H22*PNL072
IH13*P715	(Later)		(Later)	1-BJM21
IH13*P715	(Later)		(Later)	1-BJM01
IH22*P008	(Later)		(Later)	1C51-S001A
IH22*P008	(Later)		(Later)	IC51-S00LC
IH22*P008	(Later)		(Later)	IC51-S00LE
IH22*P008	(Later)		(Later)	IC51-S00LF

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION - BYPASSED

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8.4.2-1 shall be bypassed continuously by an OPERABLE bypass device integral with the motor starter or only under accident conditions by an OPERABLE control device located in the main control room, as shown in Table 3.8.4.2-1.

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 not bypassed continuously by an OPERABLE integral bypass device or only under accident conditions by an OPERABLE control device, as shown in Table 3.8.4.2-1 continuously bypass the 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

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed by an OPERABLE integral bypass device continuously or only under accident conditions by an OPERABLE control device by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing and the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions:

a. At least once per 18 months for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing.

b. At least once per 92 days for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions.

c. Following maintenance on the motor starter.

TABLE 3.8.4.2-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION - BYPASSED

VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)	SYSTEM(S) AFFECTED

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION - NOT BYPASSED

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; restore the inoperable thermal overload to OPERABLE status within 30 days 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.

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION - NOT BYPASSED

VALVE NUMBER	SYSTEM(S) AFFECTED

ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.4 Two RPS electric power monitoring channels for the each inservice RPS MG set or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

a. At least once per six months by performance of a CHANNEL FUNCTIONAL TEST; and

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Over-voltage $\leq$ 132 VAC, Bus A, and $\leq$ ( ) VAC, Bus B.

   2. Under-voltage $\geq$ 108 VAC, Bus A, and $\geq$ ( ) VAC, Bus B, and

   3. Under-frequency $\geq$ 57 Hz, + 2, - 0%.

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 main hoists 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. Unless adequate shutdown margin has been demonstrated, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.#

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.

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

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to the start of CORE ALTERATIONS, and

   2. Prior to and at least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps:

   1. Prior to control rod withdrawal,

   2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours, except that:

      1. During spiral unloading, the required count rate may be permitted to be less than 3.0 cps.

      2. Prior to and during spiral loading, until sufficient fuel has been loaded to maintain at least 3.0 cps, the required count rate may be achieved by:

         a) Use of protable external source, or

         b) Loading up to 2 fuel assemblies### in cells containing inserted control rods around an SRM.

d. Verifying, within 8 hours prior to and at least once per 12 hours during, that the RPS circuitry "shorting links" have been removed (or that the rod pattern control system is OPERABLE) during:

   1. The time any control rod is withdrawn,## or

   2. Shutdown margin demonstrations.

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

###These fuel assemblies may be loaded with the SRM count rate less than 3.0 cps.

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.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the main control room and refueling platform personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the main control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the main control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of incore instrumentation and 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 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 hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 700 pounds.

b. Demonstrating operation of the overload cutoff on the frame mounted and monorail mounted auxiliary hoists when the load exceeds 500 ± 50 pounds.

c. Demonstrating operation of the uptravel mechanical stop on the frame mounted and monorail hoists when uptravel brings the top of an active fuel assembly to 8 feet, 6 inches below the water level.

d. Demonstrating operation of the downtravel mechanical cutoff on the main hoist when grapple hook down travel reaches 4 inches below fuel assembly handle.

e. Demonstrating operation of the slack cable cutoff on the main hoist when the load is less than 50 ± 10 pounds.

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds 485 ± 50 pounds.

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 550 ± 50 pounds.

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE, TRANSFER AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 700 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage or upper containment fuel pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage or upper containment fuel pools.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 The fuel building crane physical stops which prevent crane travel with loads in excess of 700 pounds over fuel assemblies in the spent fuel storage and lower transfer pools shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.

4.9.7.2 The reactor building polar crane loads shall be verified to weigh less than or equal to 700 pounds before travel over fuel assemblies in the upper transfer and containment fuel pools.

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage and upper containment fuel pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage or upper containment fuel pools.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage or upper containment fuel pool areas, as applicable after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage and upper containment fuel pools shall be determined to be at least at its minimum required depth at least once per 7 days.

REFUELING OPERATIONS

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITION 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

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.

REFUELING OPERATIONS

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations shall be suspended unless all control rods are inserted in the core.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

f. All fuel loading operations are suspended unless all control rods are inserted in the core.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 23 feet above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish PRIMARY CONTAINMENT INTEGRITY - SHUTDOWN within 4 hours.

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

SURVEILLANCE REQUIREMENTS

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

*The shutdown cooling loop 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 train consisting of at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 23 feet above the top of the reactor pressure vessel flange.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified 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.

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access door and floor plugs of all rooms through which the transfer system penetrates are closed and locked.

b. All access interlocks and palm switches are OPERABLE.

c. The blocking valve located in the fuel building IFTS hydraulic power unit is OPERABLE.

d. All IFTS primary and secondary carriage position and liquid level indicators are OPERABLE.

e. All keylock switches which provide IFTS access control-transfer system lockout are OPERABLE.

f. The warning lights outside of the access doors are OPERABLE.

APPLICABILITY: When the IFTS containment blank flange is removed.

ACTION:

With the requirements of the above specification not satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12.1 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, verify that:

a. All access interlocks and palm switches are OPERABLE.

b. The blocking valve in the Fuel Building IFTS hydraulic power unit is OPERABLE.

c. All IFTS primary and secondary carriage position and level indicators are OPERABLE.

d. All keylock switches which provide IFTS access control-transfer system lockout are OPERABLE.

e. The warning lights outside of the access doors are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.9.12.2 Within 1 hour prior to the startup of the IFTS, verify that no personnel are in areas immediately adjacent to the IFTS tube and that the access door and floor plugs to rooms through which the IFTS tube penetrates are closed and locked.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY/DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3, 3.6.2.1, 3.6.2.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment and drywell air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to (1)% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD PATTERN CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod pattern control system (RPCS) per Specification 3.1.4 may be suspended by means of the individual rod position bypass switches for the following tests:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RPCS is OPERABLE per Specification 3.1.4.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RPCS are bypassed, verify:

a. Within 8 hours prior to bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed, that movement of the control rods from 75% ROD DENSITY to the RPCS low power setpoint is limited to the established control rod sequence for the specified test, and

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

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 per Specification 3.9.2 with the RPS circuitry "shorting links" removed.

b. The rod pattern control system OPERABLE per Specification 3.1.4, 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 "continuous withdrawal" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2, with the RPS circuitry "shorting links" removed.

b. The rod pattern control system OPERABLE, 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.

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation with matched flow may be suspended for up to 24 hours for the performance of:

a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or

b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

a. With the above specified time limit exceeded, insert all control rods.

b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified 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.

SPECIAL TEST EXCEPTIONS

3/4.10.5 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.5 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

4.10.5 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 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1.1.1-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.1.

Table 4.11.1.1.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Batch Waste Release (Liquid Radwaste Recovery Sample Tanks[b])	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	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 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,

TABLE NOTATION
continued

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

b - A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c - The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.12.

d - A composite sample is one in which the quantity of liquid sampled is proportional to the quanitity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

LIQUID RADWASTE TREATMENT SYSTEM
LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent, to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any temporary, unprotected outdoor tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above temporary unprotected outdoor tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above temporary unprotected outdoor tank 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.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE
LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be withing the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis 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. Offgas Treatment System (Pretreatment)*	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE	P Each PURGE[c] Grab Sample	P Each PURGE[c]	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C.1. Main Plant Exhaust Duct	M[c] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C.2. Fuel Building Ventilation Exhaust Duct	M[d] Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C.3. Radwaste Building Ventilation Exhaust Duct	M Grab Sample	M	Principal Gamma Emitters[b]	1×10^{-4}
D. All Release Types as listed in A, B, C above.	Continuous[e]	W[f] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[e]	W[f] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}

TABLE 4.11.2.1.2-1 (Continued)

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (µCi/ml)
D. (Continued)	Continuous[e]	Q Composite Particulate Sample	SR-89, SR-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

*If the plant uses storage tanks, each tank shall be sampled prior to release and the sample analyzed prior to release.

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 <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a</u> <u>posteriori</u> (after the fact) limit for a particular measurement.

TABLE 4.11.2.1.2-1 (Continued)

TABLE NOTATION

b - The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.9.

c - Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER within one hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor or 3.

d - Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

e - The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

f - Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

DOSE - NOBLE GASES
LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) 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.

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 radionuculides in particulate form with half-lives greater than 8 days in gaseous effluents released, to areas at and beyond the SIDE 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 radionuculides 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 Specification 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.

GASEOUS RADWASTE TREATMENT
LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4 The readings of the relevant instruments shall be checked every 12 hours when the main condenser air ejector is in use to ensure that the gaseous radwaste treatment system is functioning.

VENTILATION EXHAUST TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) would exceed 0.3 mrem to any organ in a 31 day period.

APPLICABILITY: At all times other than when the VENTILATION EXHAUST TREATMENT system is undergoing routine maintenance.

ACTION:

a. With gaseous waste being discharged from the ventilation exhaust ducts without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 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, when the ventilation exhaust treatment system is not in use.

EXPLOSIVE GAS MIXTURE
LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser offgas treatment system is in operation.

ACTION:

a. With the concentration of hydrogen or oxygen 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 whenever the main condenser evacuation system is in operation with the hydrogen monitor(s) required in compliance with Table 3.3.7.11-1 of Specification 3.3.7.11.

MAIN CONDENSER
LIMITING CONDITION FOR OPERATION

3.11.2.7 The release rate of the sum of the activities from the noble gases* measured prior to the holdup pipe shall be limited to less than or equal to 100 microcuries/sec per MW_t after 30 minutes decay.

APPLICABILITY: Whenever the main condenser offgas treatment system is in operation.

ACTION:

With the release rate of the sum of the activities from the noble gases* prior to the holdup pipe exceeding 100 microcuries/sec per MW_t after 30 minutes decay, restore release rate to within its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases prior to the holdup pipe shall be continuously monitored in accordance with Specification 3.3.7.11.

4.11.2.7.2 The release rate of the sum of the activities from the noble gases* measured prior to the holdup pipe shall be determined 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 prior to the holdup pipe.

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by OFFGAS PRETREATMENT Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

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*Gamma scintillation detectors are used to measure the Kr-85m, 87, 88 and Xe-133, 133m, 135, 138 contribution after 30 minutes decay.

3/4.11.3 SOLID RADIOACTIVE WASTE
LIMITING CONDITION FOR OPERATIO"

3.11.3 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

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

SURVEILLANCE REQUIREMENTS

4.11.3 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from any temporary unprotected outdoor storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4, Action 2.

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 6.9.1.6, 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 Specification 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant 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

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.9, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semi-annual Radioactive Effluent Release Report and 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]	40 routine monitoring stations (DR1-DR40) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in eath meterological sector in the general area of the SITE BOUNDARY (DR1-DR16); an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR17-DR32); the balance of the stations (DR33-DR40) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 SITE BOUNDARY locations, in different sectors, of the highest calculated annual average groundlevel D/Q.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
	1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-water level D/Q.		filter change;[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
	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]	1 sample upstream (Wa1) 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.
c. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value (Wd1).	Semiannually	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	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	Semimonthly when animals are on pasture, monthly at other times	Gamms isotopic[e] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.

TABLE 3.12.1-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
a. Milk (cont'd)	animals in each of 3 areas (Ia1-Ia3) betwen 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr.		
	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	1 sample of each of three 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.
	1 sample of each of three species in areas not influenced by plant discharge (Ib1 - Ib__).		
c. Food Products	1 sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged (Ic1 - Ic__).	At time of harvest[j]	Gamma isotopic analyses[e] on edible portion.
	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-Ic13).	Monthly when available	Gamma isotopic[e] and I-131 analysis.

TABLE 3.12.1-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Continued	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed (Ic20 - Ic23).	Monthly when available.	Gamma isotopic[e] and I-131 analysis.

TABLE 3.12.1-1 (Continued)

TABLE NOTATION

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 6.9.1.6. 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 environment monitoring program. Pursuant to Specification 6.9.1.9, 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 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

c - The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

d - Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12.1-1 (Continued)

TABLE NOTATION

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 samples is porportional 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 turborous 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. If no drinking water pathway exists, a value of 30,000 pCi/ℓ may be used.

TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b,c]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Nb-95	15					
Zr-95	30					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
La-140	15			15		
Ba-140	60			60		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.

TABLE 4.12.1-1 (Continued)

TABLE NOTATION

a - This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

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 distintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12.1-1 (Continued)

TABLE NOTATION

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 posterori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be 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 6.9.1.6.

d - LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

3/4.12.2 LAND USE CENSUS
LIMITING CONDITION FOR OPERATION (Continued)

3.12.1 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 Seminannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.9.

b. With land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.12, identify the new location(s) in the next Seminannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

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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, 4c shall be followed, including analysis of control samples.

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 6.9.1.6.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 requires two main control room air conditioning subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, 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 3.6.7.1 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.

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 15.0 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than (1.06). The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than (3) inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 ROD PATTERN CONTROLS

The rod withdrawal limiter system input power signal orginates from the first stage turbine pressure. When operating with the steam bypass valves open, this signal indicates a core power level which is less than the true core power. Consequently, near the low power setpoint and high power setpoint of the rod pattern control system, the potential exists for non-conservative control rod withdrawals. Therefore, when operating at a sufficiently high power level, there is a small probability of violating fuel Safety Limits during a licensing basis rod withdrawal error transient. To ensure that fuel Safety Limits are not violated, this specification prohibits control rod withdrawal when a biased power signal exists and core power exceeds the specified level.

Control rod withdrawal and insertion sequences are established to assure that the maximum 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 RPCS to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

CONTROL ROD PROGRAM CONTROLS (Continued)

The RPCS provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section (15.__) of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RPCS is also designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during higher power operation.

A dual channel system is provided that, above the low power setpoint, restricts the withdrawal distances of all non-peripheral control rods. This restriction is greatest at highest power levels.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately (90 to 120) minutes. A minimum available quantity of 3542 gallons of sodium pentaborate solution containing a minimum of 4246 lbs. of sodium pentaborate is required to meet a shutdown requirement of 3% $\Delta k/k$. There is an additional allowance of (150) ppm in the reactor core to account for imperfect mixing and the filling of other piping systems connected to the reactor vessel. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted. 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.

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

CONTROL ROD PROGRAM CONTROLS (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, 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.

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, 3.2.1-3, 3.2.1-4 and 3.2.1-5.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4 and 3.2.1-5 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram trip setpoint and the flow biased neutron flux-upscale control rod block trip setpoints 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 CMFLPD indicates a peak power distribution to ensure than an LHGR transient would not be increased in degraded conditions.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... 3015 Mwt* which corresponds to 105% of rated steam flow

Vessel Steam Output .... 13.08×10^6 lbm/hr which corresponds to (105)% of rated steam flow

Vessel Steam Dome Pressure.............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 2.2 ft^2.

b. Small Breaks 0.1 ft^2.

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.20

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3.3 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.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients. For any abnormal operating transient analysis 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 power-flow map of Figure B 3/4 2.3-1 defines the analytical basis for generation of the MCPR operating limits.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2/15._ that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154[3] and the program used in non-pressurization events is described in NEDO-10802[2]. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149[4] The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

MINIMUM CRITICAL POWER RATIO (Continued)

The $MCPR_p$s are established to protect the core from plant transients other than core flow increases, including the localized event such as rod withdrawal error. The $MCPR_p$s were calculated based upon the most limiting transient at the given core power level.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

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.

Power - Flow Operating Map

Figure B 3/4 2.3-1

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of four logic channels. The logic channels A(A1) and C(A2) comprise one trip system and the logic channels B(B1) and D(B2) comprise the other trip system for determining compliance with technical specifications. Placement of either logic channel of a trip system in the tripped condition places the trip system in the tripped condition. The trip systems as defined above are independent of each other. There are usually four instrument channels (one in each logic channel) to monitor each parameter. The tripping of a logic channel in each trip system will result in a reactor scram.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY 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 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable 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.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 140 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

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, Rod Pattern Control System, Section 3/4.2, Power Distribution Limits and Section 3/4.3, Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are 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 and 64.

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.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to 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 Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to 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 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.

3/4.3.7.9 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or 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, "LoosePart Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent 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)

3/4.3.7.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided to initiate action of the containment ventilation system and the feedwater system/main turbine trip system. The containment ventilation system provides emergency containment heat removal as described in Bases 3/4.6.3. The feedwater system/main turbine trip system is initiated in the event of failure of the feedwater controller under maximum demand.

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 loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 100°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 9 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 4 SRVs operating in the relief mode and 5 SRVs operating in the safety mode is acceptable.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

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.

3/4.4.4 CHEMISTRY (Continued)

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 (4.9) of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, phosphorus content and copper content of the material in question, can be predicted using Bases 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, 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 1.

The pressure-temperature limit lines shown in Figures 3.4.6-1, curves C, and C', and A and A', for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 77 Edition and Addenda through Summer 1978.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and 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

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	RT_{NDT} °F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	RT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS

RIVER BEND UNIT 1

BASES TABLE B 3/4.4.6-1

REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	P(%)	HIGHEST STARTING RT_{NDT}(°F)	MAX. * RT_{NDT}(°F)	AVG. UPPER SHELF (FT-LBS)	MAX. EOL RT_{NDT}(°F)
Plate	SA-533 GR B CL.1	C3138-2	0.08	0.012	+9	48	79	+57
Weld	SHELL COURSE No.2 Vertical Seam 3	492L4871/ A421B27AF	0.03	0.020	-50	80	130	+30

NOTE:* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT}(°F)
Shell Ring	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Bottom Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Dome	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Torus	SA 533 GrB Cl.1	ALL HEATS	+10
Top Head Flange	SA 508 Cl.2	ALL HEATS	+10
Vessel Flange	SA 508 Cl.2	ALL HEATS	+10
Feedwater Nozzle	SA 508 Cl.2	ALL HEATS	-20
Weld	LOW ALLOY STEEL	ALL HEATS	-20
Closure Studs	SA 540 GRADE B23 or B24	ALL HEATS	ISAE Section III, NB2300 Requirement of 45 Ft-lbs 25 mils lateral expansion at + 10°F T

Neutron Fluence, n/cm^2 (E>1MEV), x10-18

Service Life (Years*)

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

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI system, together with the LPCS system, 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.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1177 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

ECCS-OPERATING and SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 467/1400/4900 gpm at differential pressures of 1177/1147/200 psid. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety-relief valves although the safety analysis only takes credit for six valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The supression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2 or 3 is required by Specification 3.6.3.1.

SUPPRESSION POOL (Continued)

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITION 4 or 5.

In OPERATIONAL CONDITIONS 4 and 5 the suppression pool minimum required water volume is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume and vortex prevention plus a (2' 4") safety margin for conservatism.

BASES

3/4.6.1 CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 6.31 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 to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV POSITIVE LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage control system will prevent untreated leakage from the MSIV's when isolation of the primary system and containment is required.

3/5.6.1.5 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 6.31 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 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment to secondary containment differential pressure ensure that the containment peak pressure of 6.31 psig does not exceed the design pressure of 15.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of + 0.6 psid or the differential at which water would overflow the wier wall into the drywell of 0.58 psid. The limit of - 0.1 to + 1.5 psid for initial positive containment to secondary containment pressure will limit the containment pressure to 6.31 psid which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on primary containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 185°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND PRIMARY CONTAINMENT PURGE SYSTEM

The (20) inch (drywell and) containment 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) inch valves cannot be inadvertently opened, 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) containment purge lines is restricted to the (6) inch purge supply and exhaust isolation valves since, unlike the 42-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 integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.1.9 PENETRATION VALVE LEAKAGE CONTROL SYSTEM

The OPERABILITY of the penetration valve leakage control system is required to meet the restrictions on overall containment leak rate assumed in the accident analyses. The Surveillance Requirements for determining OPERABILITY are consistent with Appendix "J" of 10 CFR 50.

3/4.6.2 DRYWELL

3/4.6.2.1 DRYWELL INTEGRITY

Drywell integrity ensures that the steam released for the full spectrum of drywell pipe breaks is condensed inside the primary containment either by the suppression pool or by the primary containment ventilation system unit coolers. By utilizing the suppression pool as a heat sink, energy released to the containment is minimized and the severity of the transient is reduced.

3/4.6.2.2 DRYWELL BYPASS LEAKAGE

The limitation on drywell bypass leakage rate is based on having at least one containment ventilation system unit cooler OPERABLE. It ensures that the maximum leakage which could bypass the suppression pool during an accident would not result in the containment exceeding its design pressure of 15.0 psig. The integrated drywell leakage value is limited to 10% of the design drywell leakage rate.

The limiting case accident is a very small reactor coolant system break which will not automatically result in a reactor depressurization. The long term differential pressure created between the drywell and containment will result in a significant pressure buildup in the containment due to this bypass leakage.

3/4.6.2.3 DRYWELL AIR LOCKS

The limitations on closure for the drywell air locks are required to meet the restrictions on DRYWELL INTEGRITY and the drywell leakage rate given in Specifications 3.6.2.1 and 3.6.2.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in the air lock is required to maintain the integrity of the drywell.

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that the drywell peak pressure of 18.6 psid does not exceed the design pressure of 25.0 psid and that the containment peak pressure of 6.31 psig does not exceed the design pressure of 15.0 psig during LOCA conditions. The maximum external drywell pressure differential is limited to + 0.1 psid, well below the 0.58 psid at which suppression pool water will be forced over the wier wall and into the drywell. The limit of 0.5 psid for initial positive drywell to containment pressure will limit the drywell pressure to 18.9 psid which is less than the design pressure and is consistent with the safety analysis.

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that peak drywell temperature does not exceed the design temperature of 330 °F during LOCA conditions and is consistent with the safety analysis.

3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and containment pressure will not exceed the design pressure of 25 psig and 15 psig, respectively, during primary system blowdown from full operating pressure.

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from 1060 psig. Using conservative parameter inputs, the maximum calculated containment pressure during and following a design basis accident is below the containment design pressure of 15 psig. Similarly the drywell pressure remains below the design pressure of 25 psig. The maximum and minimum water volumes for the suppression pool are 138,851 cubic feet and 136,146 cubic feet, respectively. These values include the water volume of the containment pool, horizontal vents, and weir annulus. Testing in the Mark III Pressure Suppression Test Facility and analysis have assured that the suppression pool temperature will not rise above 185°F for the full range of break sizes.

Should it be necessary to make the suppression pool inoperable, this shall only be done as specified in Specification 3.5.3.

Experimental data indicates that effective steam condensation without excessive load on the containment pool walls will occur with a quencher device and pool temperature below 200°F during relief valve operation. Specifications have been placed on the envelope of reactor operating conditions to assure the bulk pool temperature does not rise above 185°F in compliance with the containment structural design criteria.

In addition to the limits on temperature of the suppression pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

The containment ventilation system consists of three 100% capacity unit coolers, two of which are safety related. Each of these two unit coolers provides independent 100% heat removal capacity in case of steam bypass of the supression pool. The turbulence caused by the spray system aids in mixing the containment air volume to maintain a homogeneous mixture for H_2 control.

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the system is to ensure containment integrity following a LOCA by preventing excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F

DEPRESSURIZATION SYSTEMS (Continued)

considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related component systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the standby service water is accomplished in the RHR heat exchangers.

3/4.6.4 PRIMARY CONTAINMENT AND DRYWELL ISOLATION VALVES

The OPERABILITY of the 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 to 10 CFR 50. Containment and drywell isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The operability of the drywell isolation valves ensures that the drywell atmosphere will be directed to the suppression pool for the full spectrum of pipe breaks inside the drywell and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Since the allowable value of drywell leakage is so large, individual drywell penetration leakage is not measured. By checking valve operability on any penetration which could contribute a large fraction of the design leakage, the total leakage is maintained at less than the design value.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Shield Building, Annulus, Auxiliary Building and Fuel Building with associated structures provide secondary containment during normal operation when the containment is sealed and in service. At other times, the containment may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the Shield Building, Annulus and Auxiliary Building with the standby gas treatment system and in the Fuel Building with the ventilation system charcoal filtration subsystem once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the standby gas treatment systems and the ventilation charcoal filtration subsystems ensure that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of these systems and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of these systems with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters.

3/4.6.6 ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the containment below its flammable limit during post-LOCA conditions. Either containment hydrogen recombiner system is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. The containment/drywell hydrogen mixing systems are provided to ensure adequate mixing of the containment and drywell atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Two 100% containment/drywell mixing systems are the primary means of H_2 control within the drywell, exhausting hydrogen produced following a LOCA into the containment volume. Hydrogen generated from the metal-water reaction and radiolysis is assumed to evolve to the drywell atmosphere and form a homogenous mixture through natural forces and mechanical turbulence ECCS pipe break flow. The containment/drywell hydrogen mixing system recirculates drywell atmosphere within the containment diluting the drywell hydrogen concentration.

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.

ATMOSPHERE CONTROL (Continued)

The operability of the containment and drywell hydrogen igniters ensures that hydrogen combustion can be accomplished in a controlled manner following a degraded core event producing hydrogen concentrations in excess of LOCA conditions.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 STANDBY SERVICE WATER SYSTEM

The OPERABILITY of the service water system and ultimate heat sink ensure that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 MAIN CONTROL ROOM AIR CONDITIONING SYSTEM

The OPERABILITY of the main control room air conditioning system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

REACTOR CORE ISOLATION COOLING SYSTEM (Continued)

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

3/4.7.4 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event 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.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed, nominal time less 25%, may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those snubbers which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

SNUBBERS (Continued)

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample according to the expression 35 $(1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers will require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records ,i.e., newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . . . The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4 7.6 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinkler systems, Halon system and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide 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.

3/4.7.7 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.7.8 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause loss of its OPERABILITY. The temperature limits include allowance for an instrument error of ( )°F.

3/4.7.9 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis in FSAR Chapter 15.

3/4 7.10 STRUCTURAL SETTLEMENT

Structural settlement limitations are imposed and required to be verified so as to preserve the assumptions made in the static design of the major safety related structures.

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 Division I or II of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. or D.C. source. Division III supplies the high pressure core spray (HPCS) system only.

The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When diesel generator 1A or 1B is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator 1A or 1B as a source of emergency power, are also OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period diesel generator 1A or 1B 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", March 10, 1971, and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants", Revision 1, August 1977.

A.C. SOURCES, D.C. SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The surveillance requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants", February 1978, and IEEE Std 450-1975, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, 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 .015 below the manufacturer's nominal full charge specific gravity of 1.215 or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations 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 or bypassing of the motor operated valves thermal overload protection continuously or during accident conditions ensures that the thermal overload protection during accident conditions will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY or bypassing of the thermal overload protection continuously and or during accident conditions are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.

3/4.9.1 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 platform personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each hoist has sufficient load capacity for handling fuel assemblies and/or control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE, TRANSFER AND UPPER CONTAINMENT FUEL POOLS

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the pools ensures that in the event this load is dropped 1) the activity release will be limited to that contained in 123 fuel rods, 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 AND UPPER CONTAINMENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE and in operation or that an altenate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event 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.

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

The purpose of the inclined fuel transfer system specification is to control personnel access to those potentially high radiation areas immediately adjacent to the system and to assure safe operation of the system.

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY/DRYWELL INTEGRITY

The requirements for PRIMARY CONTAINMENT INTEGRITY and DRYWELL INTEGRITY are not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD PATTERN CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance 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 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.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials

3/4.11.1.2 DOSE (Continued)

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 requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tank listed in this Specification include those temporary unprotected outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that te dose at any time at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of

3/4.11.2.1 DOSE RATE (Continued)

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 effluent samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 20 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and 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.

3/4.11.2.4 AND 3/4.11.2.5 GASEOUS RADWASTE TREATMENT AND VENTILATION EXHAUST TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT (OFF GAS) SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4 11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the main condenser offgas treatment system is maintained below the flammability limits of hydrogen. (Automatic control features are included in the system to prevent the hydrogen concentration from reaching this flammability limit. These automatic control features include isolation of the source of hydrogen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limit.) Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criteria 3 and 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MAIN CONDENSER

Restricting the release rate of the sum of the activities from the noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effleuent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12.1-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Handford Company Report ARH-SA-215 (June 1975).

3/4/12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at any beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

3/4.12/3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environment monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

UNRESTRICTED AREA AND SITE BOUNDARY FOR GASEOUS EFFLUENTS AND FOR LIQUID EFFLUENTS

5.1.3 The UNRESTRICTED AREA AND SITE BOUNDARY for gaseous effluents and for liquid effluents shall be as shown in Figure 5.1.3-1. The gaseous effluent release points are shown in Figure 5.1.1-1.

5.2 CONTAINMENT

PRIMARY CONTAINMENT

5.2.1 The primary containment is a steel structure composed of a vertical right cylinder and a torispherical dome. Inside and at the bottom of the containment is a reinforced concrete drywell composed of a vertical right cylinder and a steel head. Primary containment contains a approximately 20 feet deep water filled suppression pool connected to the drywell through a series of horizontal vents. The primary containment has a minimum net free air volume of 1,190,000 cubic feet. The drywell has a minimum net free air volume of 236,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The containment and drywell are designed and shall be maintained for:

a. Maximum internal pressure:
   1. Drywell 25 psig.
   2. Containment 15 psig.

b. Maximum internal temperature:
   1. Drywell 330°F.
   2. Suppression pool 185°F.

(c. Maximum external to internal differential pressure:
   1. Drywell 20 psid.
   2. Containment 0.6 psid.)

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the shield building, the auxiliary building and the fuel building. Secondary containment has a minimum free volume of 2,278,000 cubic feet.

This figure shall consist of a map of the site area and provide at a minimum, the information described in Section (2.1.2) of the FSAR and meteorolgical tower location.

EXCLUSION AREA

FIGURE 5.1.1-1

This figure shall consist of a map of the site area showing the Low Population Zone boundary. Features such as towns, roads and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.

LOW POPULATION ZONE

FIGURE 5.1.2-1

UNRESTRICTED AREA AND SITE BOUNDARY

Figure 5.1.3-1

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 624 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 1.70 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 145 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143.7 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. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

   3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 16,000 cubic feet at a nominal steam dome saturation temperature of 549°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

5.6.1.1 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, including all calculational uncertainties and biases as described in Section 9.1.2 of the FSAR.

b. A nominal 7 inch center to center storage spacing in the low density storage racks.

c. A nominal 6.28 inch center to center storaged spacing, with neutron absorbers consisting of 0.77 inch of stainless steel and 0.020 gram per square centimeter of B-10 between an assembly and each of its neighbors, in the high density storaged racks.

The storage of spent fuel in the upper containment fuel storage pool is prohibit during normal operation.

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 0.95 when dry or flooded with unborated water.

5.6.1.3 The new fuel storage vault racks are designed and shall be maintained with:

a. A normal 7.0 inch center-to-center spacing within rows and a nominal 12.25 inch center-to-center spacing between rows.

b. The K_{eff} for new fuel stored dry in the new fuel storage vault racks shall not exceed 0.95 when dry or flooded with unborated water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 95'.

CAPACITY

5.6.3 The spent fuel storage pool in the fuel building is designed and shall be maintained with a storage capacity limited to no more than 2680 fuel assemblies.

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
	80 step change cycles	Loss of feedwater heaters
	180 reactor trip cycles	100% to 0% of RATED THERMAL POWER
	40 hydrostatic pressure or leak tests	Pressurized to $\geq$ 930 psig and $\leq$ 1250 psig

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor or during his absence from the control room, a designated individual shall be responsible for the control room command function. A management directive to this effect, signed by the Senior Vice President - River Bend Nuclear Group shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2.1-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2.2-1 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room;

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site fire brigade of at least five members shall be maintained on site at all times*. The fire brigade shall not include the Shift Supervisor, the Shift Technical Advisor, the Control Operating Foreman, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and

*The Radiation Protection Technician and fire brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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, or major unit modifications, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or 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 Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

This figure shall show the organizational structure and lines of responsibility for the offsite groups that provide technical and management support for the unit. The organizational arrangement for performing and monitoring quality assurance activities shall also be indicated.

FIGURE 6.2.1-1

OFFSITE ORGANIZATION

This figure shall show the organizational structure and lines of responsibility for the unit staff. Positions to be staffed by licensed personnel shall be indicated. The organizational arrangement for performing and monitoring quality assurance activities shall also be indicated.

FIGURE 6.2.2-1

UNIT ORGANIZATION

T\BLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

(Also for multiple unit site with only one unit licensed)

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITION 1, 2, or 3	CONDITION 4 or 5
SS	1	1
COF	1	None
NCO	2	1
NEO	2	1
STA	1	None

TABLE NOTATION

SS - Shift Supervisor with a Senior Operator license on Unit 1.
COF - Control Operating Foreman with a Senior Operator license on Unit 1.
NCO - Nuclear Control Operator with an Operator license on Unit (1).
YEO - Nuclear Equipment Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the shift crew composition may be one less than the minimum requirements of Table 6.2.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 1, 2 or 3, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in OPERATIONAL CONDITION 4 or 5, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Director - Nuclear Plant Engineering.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located onsite. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Director - Nuclear Plant Engineering.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the (Radiation Protection Chemistry 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.

*Not responsible for sign-off function.

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 5.5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 FACILITY REVIEW COMMITTEE (FRC)

FUNCTION

6.5.1.1 The FRC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The FRC shall be composed of the:

Chairman:	Assistant Plant Manager-Operations
Member:	Assistant Plant Manager-Services
Member:	Operations Supervisor
Member:	General Maintenance Supervisor
Member:	Radiation Protection/Chemistry Supervisor
Member:	Reactor Engineering Supervisor

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the FRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in FRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The FRC shall meet at least once per calendar month and as convened by the FRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the FRC necessary for the performance of the FRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including no more than two alternates.

RESPONSIBILITIES

6.5.1.6 The FRC shall be responsible for:

a. Review of (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 Plant Manager 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 Senior Vice President - RBNG and to the (Company Nuclear Review and Audit 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 Plant Manager or the (Company Nuclear Review and Audit Group);

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the (Company Nuclear Review and Audit Group); and

j. Review of the Emergency Plan and implementing procedures and submittal of the recommended changes to the (Company Nuclear Review and Audit Group).

6.5.1.7 The FRC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6.a. through d. prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6.a. through e. constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the (Vice President - Nuclear Operations) and the (Company Nuclear Review and Audit Group) of disagreement between the FRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The FRC shall maintain written minutes of each FRC meeting that, at a minimum, document the results of all FRC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Senior Vice President - RBNG and the (Company Nuclear Review and Audit Group).

6.5.2 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.2.1 The NRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Quality assurance practices.

The NRB shall report to and advise the Senior Vice President - RBNG on those areas of responsibility in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The NRB shall be composed of the:

Chairman:	Senior Vice President - External Affairs
Member; Alt. Chairman	Vice President - RBNG
Member:	Vice President-Administration
Member:	Manager-Design Engineering, Technical Services Department
Member:	Manager-Engineering, Nuclear Fuels, and Licensing
Member:	Plant Manager
Member:	Manager-Quality Assurance
Member:	Director-Nuclear Licensing
Member:	Director-Nuclear Plant Engineering
Member:	Director-Nuclear Fuels Design and Safety Analysis

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NRB activities at any on[illegible]ime.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.2.5 The NRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NRB necessary for the performance of the NRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four NRB members including no more than two alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NRB shall review:

a. The safety evaluations for (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS:

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the NRB.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the NRB. 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 qualified offsite licensee personnel or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least every third year; and

g. Any other area of unit operation considered appropriate by the NRB or the Senior Vice President - RBNG.

h. The Emergency Plan and implementing procedures at least once per 12 months.

i. The Security Plan and implementing procedures at least once per 12 months.

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

l. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

m. The performance of activities by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15 at least once per 12 months.

RECORDS

6.5.2.9 Records of NRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NRB meeting shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Senior Vice President - RBNG within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Senior Vice President - RBNG and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.73 and

b. Each REPORTABLE EVENT shall be reviewed by the FRC and the results of this review shall be submitted to the NRB and the Senior Vice President - RBNG.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President - RBNG, and the NRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NRB, and the Senior Vice President - RBNG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737.

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.

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the FRC and shall be approved by the Plant Manager 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 the unit affected; and

c. The change is documented, reviewed by the FRC, and approved by the Plant Manager 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 HPCS, LPCS, RHR, RCIC, process sampling and standby gas treatment systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

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.

STARTUP REPORT (Continued)

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. Documentation of all challenges to safety/relief valves.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience (including documentation of all challenges to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

*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.

**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate), with preoperational studies, operational controls and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of all radiological environmental samples and of all environmental radiation measurements taken during the report period pursuant to the locations specified in the tables and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor plant; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 4.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

SEMIANNUAL EFFLUENT RELEASE REPORT***

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

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*A single submittal may be made for a multiple unit station.

**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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction and atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind directon, atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 5.1.3-1a and 5.1.3-1b) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters of the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each type of solid waste (as defined in 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),

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*In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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 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.

i. Records of analyses required by the radiological environmental monitoring program.

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.

RECORD RETENTION (Continued)

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

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

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

h. Records of 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 FRC and the NRB.

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

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

*Radiation protection personnel or personnel escorted by radiation protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

c. A radiation protection 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 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Control Operating Foreman on duty and/or the radiation protection supervision. Doors shall remain locked except during periods of 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 mrem* 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.

*Measurement made at 18 inches from source of radioactivity.

6.13 PROCESS CONTROL PROGRAM (PCP)

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

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRC.

2. Shall become effective upon review and acceptance by the FRC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and data box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRC.

2. Shall become effective upon review and acceptance by the FRC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the FRC. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59.

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change, which shows 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;

   e. An evaluation of the change, which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the FRC.

2. Shall become effective upon review and acceptance by the FRC.