NUREG-1253

Technical Specifications
Nine Mile Point Nuclear Station, Unit No. 2

Docket No. 50-410

Appendix "A" to
License No. NPF-69

Issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

July 1987

UNITED STATES NUCLEAR REGULATORY COMMISSION

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1.0 DEFINITIONS

The following terms are defined so that these specifications may be uniformly interpreted. The defined terms appear in capitalized type 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 so 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.

CHANNEL FUNCTIONAL TEST

1.6 (Continued)

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps so 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 fuel assembly operating power.

DOSE EQUIVALENT I-131

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ 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, expressed 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 RESPONSE TIME

1.12 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

EMERGENCY CORE COOLING SYSTEM RESPONSE TIME

1.12 (Continued)

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 so that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 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 so that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 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

IDENTIFIED LEAKAGE

1.18 (Continued)

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

ISOLATION SYSTEM RESPONSE TIME

1.19 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 so that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern that results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MINIMUM CRITICAL POWER RATIO (MCPR).

LINEAR HEAT GENERATION RATE

1.21 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

1.22 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 so that the entire logic system is tested.

MEMBER(S) OF THE PUBLIC

1.23 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the Nine Mile Point Nuclear Station and James A. FitzPatrick Nuclear Power Plant. This category does not include employees of Niagara Mohawk Power Corporation, the New York State Power Authority, their 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 Nine Mile Point Nuclear Station and James A. FitzPatrick Nuclear Power Plant.

MILK SAMPLING LOCATION

1.24 A MILK SAMPLING LOCATION is a location where 10 or more head of milk animals are available for the collection of milk samples.

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

1.26 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses that result from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.27 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 primary containment penetrations required to be closed during accident conditions are either:

PRIMARY CONTAINMENT INTEGRITY

1.31 (Continued)

1. Capable of being closed by an OPERABLE primary 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 3.6.3-1 of Specification 3.6.3.

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

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

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

e. The suppression pool is in compliance with the requirements of Specification 3.6.2.1.

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

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

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

RATED THERMAL POWER

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 (Continued)

until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping, or total steps so that the entire response time is measured.

REPORTABLE EVENT

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

ROD DENSITY

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

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor building and auxiliary bay penetrations required to be closed during accident conditions are either:

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

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

b. All auxiliary bay hatches are closed and sealed.

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

d. At least one door in each access to the reactor building and auxiliary bays is closed except during normal entry and exit.

e. The sealing mechanism associated with each reactor building and auxiliary bay penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

f. The pressure within the reactor building and auxiliary bays is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.39 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

SHUTDOWN MARGIN

1.39 (Continued)

assumed to be fully withdrawn and the reactor is in the shutdown condition, cold (i.e., 68°F), and xenon free.

SITE BOUNDARY

1.40 The SITE BOUNDARY shall be that line around the Nine Mile Point Nuclear Station beyond which the land is not owned, leased, or otherwise controlled by the Niagara Mohawk Power Corporation or the New York State Power Authority.

SOLIDIFICATION

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

SOURCE CHECK

1.42 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

1.43 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TURBINE BYPASS SYSTEM RESPONSE TIME

1.45 The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two time intervals:

a. Time from initial movement of the main turbine stop valve or control valve until 80% of turbine bypass capacity is established, and

b. the time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

   Either response time may be measured by any series of sequential, overlapping, or total steps, so that both entire response time components are measured.

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.47 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY, access to which is not controlled by the Niagara Mohawk Power Corporation or the New York State Power Authority 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

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

VENTING

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

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATIONS

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

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

TABLE NOTATIONS

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

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

† 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.

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.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 with two recirculation loop operation and shall not be less than 1.07 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.06, with two recirculation loop operation or less than 1.07 with single loop operation, the reactor vessel steam dome pressure greater than 785 psig, and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

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.

REACTOR VESSEL WATER LEVEL

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

SAFETY LIMITS REACTOR VESSEL WATER LEVEL

2.1.4 (Continued)

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.

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 VALUE
1. Intermediate Range Monitor, - Nuetron Flux - High	≤120/125 divisions of full scale	≤122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux - Upscale, Setdown	≤15% of RATED THERMAL POWER	≤20% of RATED THERMAL POWER
b. Flow-Biased Simulated Thermal Power - Upscale		
1) Flow-Biased 2) High-Flow-Clamped	≤0.66 (W-ΔW)(a) + 51%, with a maximum of <113.5% of RATED THERMAL POWER	≤0.66 (W-ΔW)(a) + 54%, with maximum of <115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - Upscale	≤118% of RATED THERMAL POWER	≤120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1037 psig	≤1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥159.3 in. above instrument zero*	≥157.8 in. above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤8% closed	≤12% closed
6. Main Steam Line Radiation - High	≤3.0 x full-power background	≤3.6 x full-power background
7. Drywell Pressure - High	≤1.68 psig	≤1.88 psig

* See Bases Figure B3/4 3-1.

(a) The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ΔW=0 for two loop operation. ΔW=5% for single loop operation.

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter/Trip Unit	≤43.4 in.	≤49.5 in.
b. Float Switch	≤48.5 in.	≤49.5 in.
9. Turbine Stop Valve - Closure	≤5% closed	≤7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥530 psig	≥465 psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

2.1 BASES FOR SAFETY LIMITS

2.1.0 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 so 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 so that the MCPR is not less than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation. MCPR greater than 1.06 for two recirculation loop operation and 1.07 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers that 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 that occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. Although 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 that would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5-psi driving head will be greater than 28×10^3 lb/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 so that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that 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 in which 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*, 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** and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958-A*. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

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

** 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	
Two Recirculation Loop Operation	2.5
Single Recirculation Loop Operation	6.0
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	
Two Recirculation Loop Operation	6.3
Single Recirculation Loop Operation	6.8
R Factor	1.5
Critical Power	3.6

* The uncertainty analysis used to establish the corewide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core. The values herein apply to both two recirculation loop operation and single recirculation loop operation, except as noted.

BASES TABLE B2.1.2-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

PARAMETER	VALUE
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 so 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 Winter 1972, which permits a maximum pressure transient of 110%, 1375 psig, of the 1250 psig reactor pressure vessel design pressure. 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 Section III of the ASME Boiler and Pressure Vessel Code, 1977 Edition, including Addenda through Summer 1977, for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressure. The design pressures are 1250 psig for suction piping and 1650 psig for discharge piping to the exit of the discharge block valve and 1550 psig for the remainder of the discharge piping to the vessel nozzles. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Addenda through Winter 1972.

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

2.2 BASES FOR LIMITING SAFETY SYSTEM SETTINGS

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 an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoints and allowable values also contain additional margin for instrument accuracy and calibration capability.

1. Intermediate Range Monitor, Neutron Flux - High

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

The most significant source of reactivity changes during the power increase is control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents has been analyzed. The results of these analyses are in Section 15.4 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 that the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shut down and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. On the basis of this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and also provides backup protection for the APRM.

2. Average Power Range Monitor

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 (Continued)

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 that result from transient operation for the case of the Fixed Neutron Flux - Upscale setpoint; i.e., for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux because of the time constants of the heat transfer associated with the fuel. For the Flow-Biased Simulated Thermal Power - Upscale 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 greater than or equal to FRTP.

3. Reactor Vessel Steam Dome Pressure - High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase during operation 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 with 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 load rejection or a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

4. Reactor Vessel Water Level - Low

The reactor vessel water level Trip Setpoint 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.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 (Continued)

5. Main Steam Line Isolation Valve - Closure

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

6. Main Steam Line Radiation - High

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

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

8. Scram Discharge Volume Water Level - High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The Trip Setpoint for each scram discharge volume is equivalent to a contained volume of approximately 24 gallons of water. This corresponds to a level indicating switch reading of 43.4 inches above an instrument zero level of elevation 263 feet 10 inches.

9. Turbine Stop Valve - Closure

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

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 (Continued)

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.

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

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection with or without coincident failure of the turbine bypass valves. The reactor protection system initiates a trip in less than 30 milliseconds* after the start of control valve fast closure when fast closure of the control valves is initiated by the fast-acting solenoid valves. 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.

11. Reactor Mode Switch Shutdown Position

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

12. Manual Scram

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

*In the analysis, 50 milliseconds is assumed for RPS relay logic response time. The analysis assumes that de-energization of the scram solenoids occurs in less than or equal to 80 milliseconds from start of Turbine Control Valve fast closure.

3/4.0 APPLICABILITY

LIMITING CONDITIONS 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 1 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.

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 three 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 specifications. 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, and 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 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 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

4.0.5 (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 CONDITIONS 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 before 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.

REACTIVITY CONTROL SYSTEMS

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITIONS 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.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITIONS 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 because it is immovable as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within 1 hour:

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

      b) Disarm the associated directional control valves* either:

         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      c) Comply with Surveillance Requirement 4.1.1.c.

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

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

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

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

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

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

CONTROL ROD OPERABILITY

LIMITING CONDITIONS FOR OPERATION

3.1.3.1.b.1 (Continued)

ACTION:

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

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

a) Electrically, or
b) Hydraulically by closing the drive water and exhaust water isolation valves.

2. If the inoperable control rod(s) is inserted, within 1 hour disarm the associated directional control valves** either:

a) Electrically, or
b) Hydraulically by closing the drive water and exhaust water isolation valves.

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

3. The provisions of Specification 3.0.4 are not applicable.

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

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

e. With any scram discharge volume vent valve(s) and/or any scram discharge volume drain valve(s) otherwise inoperable, restore at least one vent valve and one drain valve to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

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

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

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 rod worth minimizer (RWM) and rod sequence control system (RSCS), all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and
b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

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

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

a. The scram discharge volume drain and vent valves OPERABLE when the 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.

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

**The provisions of Specification 4.0.4 are not applicable for entry into Operational Condition 2 provided the Surveillance is performed within 12 hours after achieving less than or equal to 50% rod density.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITIONS FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

   1. Declare the control rod(s) with the slow insertion time 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 control rods with maximum scram insertion times in excess of 7.0 seconds.

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

CONTROL ROD AVERAGE SCRAM INSERTION TIMES

LIMITING CONDITIONS FOR OPERATION

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

POSITION INSERTED FROM FULLY WITHDRAWN	AVERAGE SCRAM INSERTION TIME (SECONDS)
45	0.43
39	0.86
25	1.93
5	3.49

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

FOUR CONTROL ROD GROUP SCRAM INSERTION TIMES

LIMITING CONDITIONS FOR OPERATION

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

POSITION INSERTED FROM FULLY WITHDRAWN	AVERAGE SCRAM INSERTION TIME (SECONDS)
45	0.45
39	0.92
25	2.05
5	3.70

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

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

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITIONS FOR OPERATION

3.1.3.5 All control rod scram accumulators shall be OPERABLE.

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

ACTION:

a. In OPERATIONAL CONDITIONS 1 or 2:

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

   a) Restore the inoperable accumulator to OPERABLE status, or
   b) Declare the control rod associated with the inoperable accumulator inoperable.

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

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

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

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

      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

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

b. In OPERATIONAL CONDITION 5*:

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

   a) Electrically, or

---

* 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.

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITIONS FOR OPERATION

3.1.3.5.b.1 (Continued)

ACTION:

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

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

a. At least once per 7 days by verifying that the indicated pressure is greater than or equal to 940 psig unless the control rod is inserted and disarmed or scrammed.

b. At least once per 18 months by:

1. Performance of a:

   a) CHANNEL FUNCTIONAL TEST of the leak detectors, and
   b) CHANNEL CALIBRATION of the pressure detectors, and verifying an alarm setpoint of greater than or equal to 940 psig on decreasing pressure.

2. Measuring and recording for up to 10 minutes that each individual accumulator check valve maintains the associated accumulator pressure above the alarm setpoint with no control rod drive pump charging water supplying the scram accumulators by closing charging water manual isolation valve V28 and depressurizing charging water header by opening valves V67 and V68.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD DRIVE COUPLING

LIMITING CONDITIONS FOR OPERATION

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

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

ACTION:

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

   1. If permitted by the rod worth minimizer (RWM) and rod sequence control system (RSCS), insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

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

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

      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 RWM or RSCS, then until permitted by the RWM and RSCS, declare the control rod inoperable, insert the control rod, and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

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

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

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

---

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

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

LIMITING CONDITIONS FOR OPERATION

3.1.3.6.b.1 (Continued)

ACTION:

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

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

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

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

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

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

CONTROL ROD POSITION INDICATION

LIMITING CONDITIONS FOR OPERATION

3.1.3.7 The control rod position indication system shall be OPERABLE.

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

ACTION:

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

   1. Determine the position of the control rod

      a) By full-out indication if rod is fully withdrawn, or

      b) By full-in indication if rod is fully inserted, or

      c) For the affected inoperable control rod position, verify one notch "out" and one notch "in" control rod indicators OPERABLE, and

      d) Verifying no control rod drift alarm at least once per 12 hours, 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 RSCS:

         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 member of the unit technical staff.

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

         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

---

* 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.

CONTROL ROD POSITION INDICATION

LIMITING CONDITIONS FOR OPERATION

3.1.3.7 (Continued)

ACTION:

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

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD DRIVE HOUSING SUPPORT

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITIONS FOR OPERATION

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

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

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

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

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

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

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

---

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

REACTIVITY CONTROL SYSTEMS

ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITIONS FOR OPERATION

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

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

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RSCS shall be demonstrated OPERABLE by:

   a. Performance of a self-test:

      1. Within 8 hours prior to each reactor startup, and
      2. Prior to movement of a control rod after rod inhibit mode automatic initiation when reducing THERMAL POWER.

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

      1. After withdrawal of the first in-sequence control rod for each reactor startup, and

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

---

* See Special Test Exception 3.10.2.

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

ROD BLOCK MONITOR

LIMITING CONDITIONS FOR OPERATION

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

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

ACTION:

a. With one RBM channel inoperable:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITIONS 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.

   3. The temperature of the pump suction piping is greater than or equal to 70°F.

---

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

SURVEILLANCE REQUIREMENTS

4.1.5 (Continued)

b. At least once per 31 days by:

   1. Verifying the continuity of the explosive charge.

   2. Determining that the available weight of sodium pentaborate is greater than or equal to 5500 lb and the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

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

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm 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 1387** psig and verifying that the relief valve does not actuate during recirculation to the test tank.

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

** Bench-tested setpoint value.

SURVEILLANCE REQUIREMENTS

4.1.5.d (Continued)

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

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

*This test shall also be performed 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.

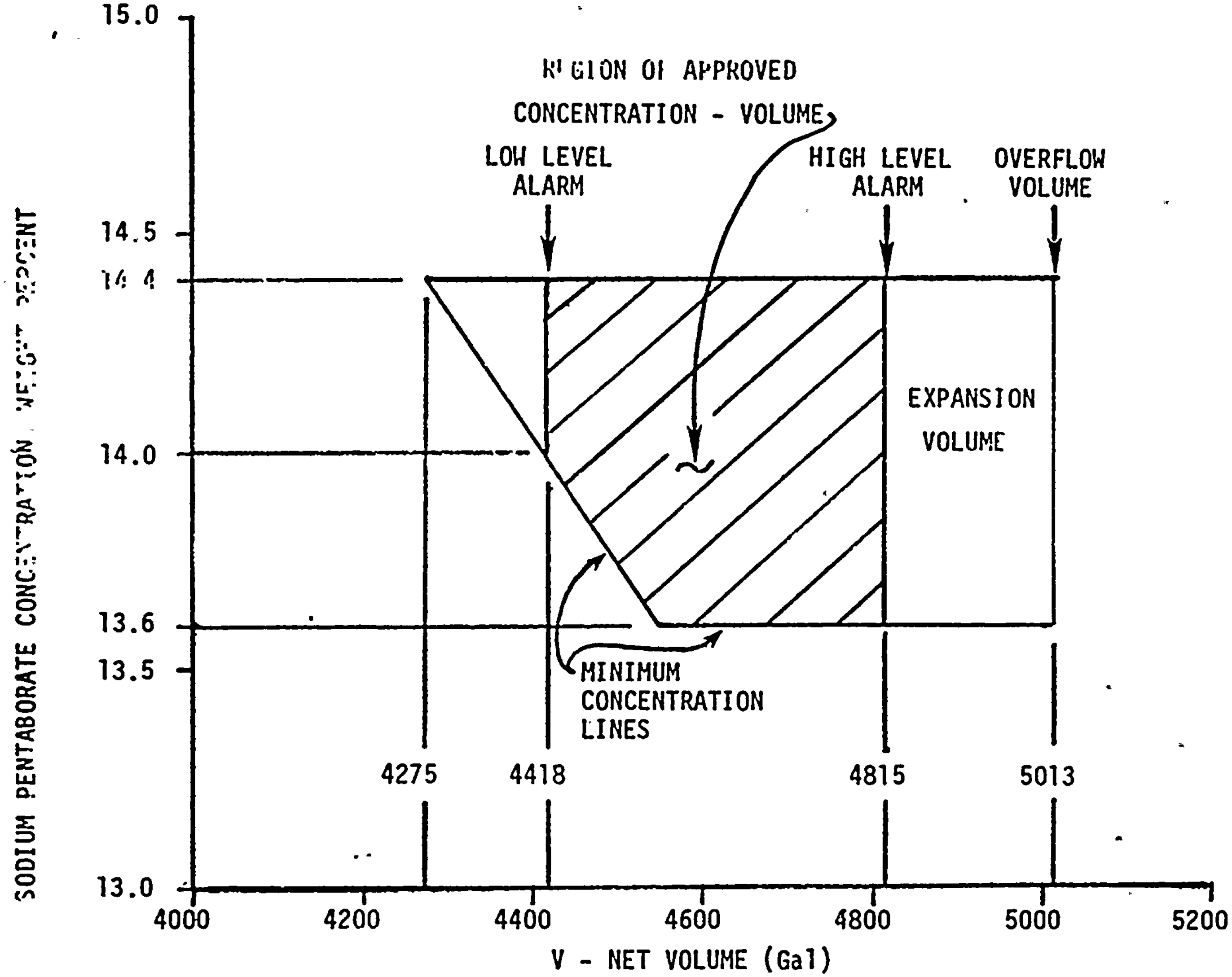

Figure 3.1.5-1 Sodium-Pentaborate Tank Volume vs. Concentration Requirements

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITIONS FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3. The limits of Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 shall be reduced to a value of 0.81 times the two recirculation loop operation limit when in single recirculation loop operation.

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

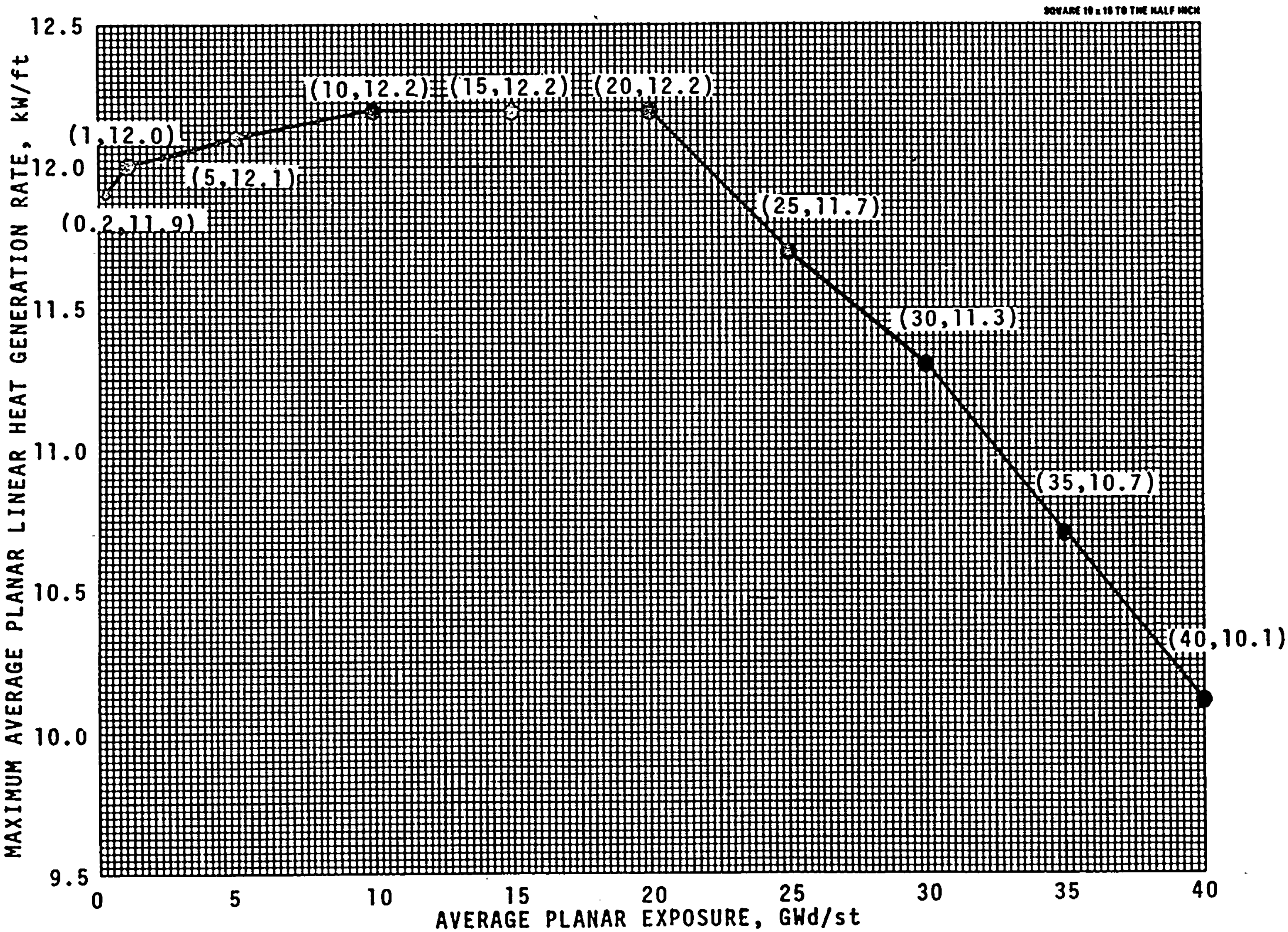

Figure 3.2.1-1 Maximum Average Planar Linear Heat Generation Rate vs. Average Planar Exposure, Fuel Type BP8CRB219

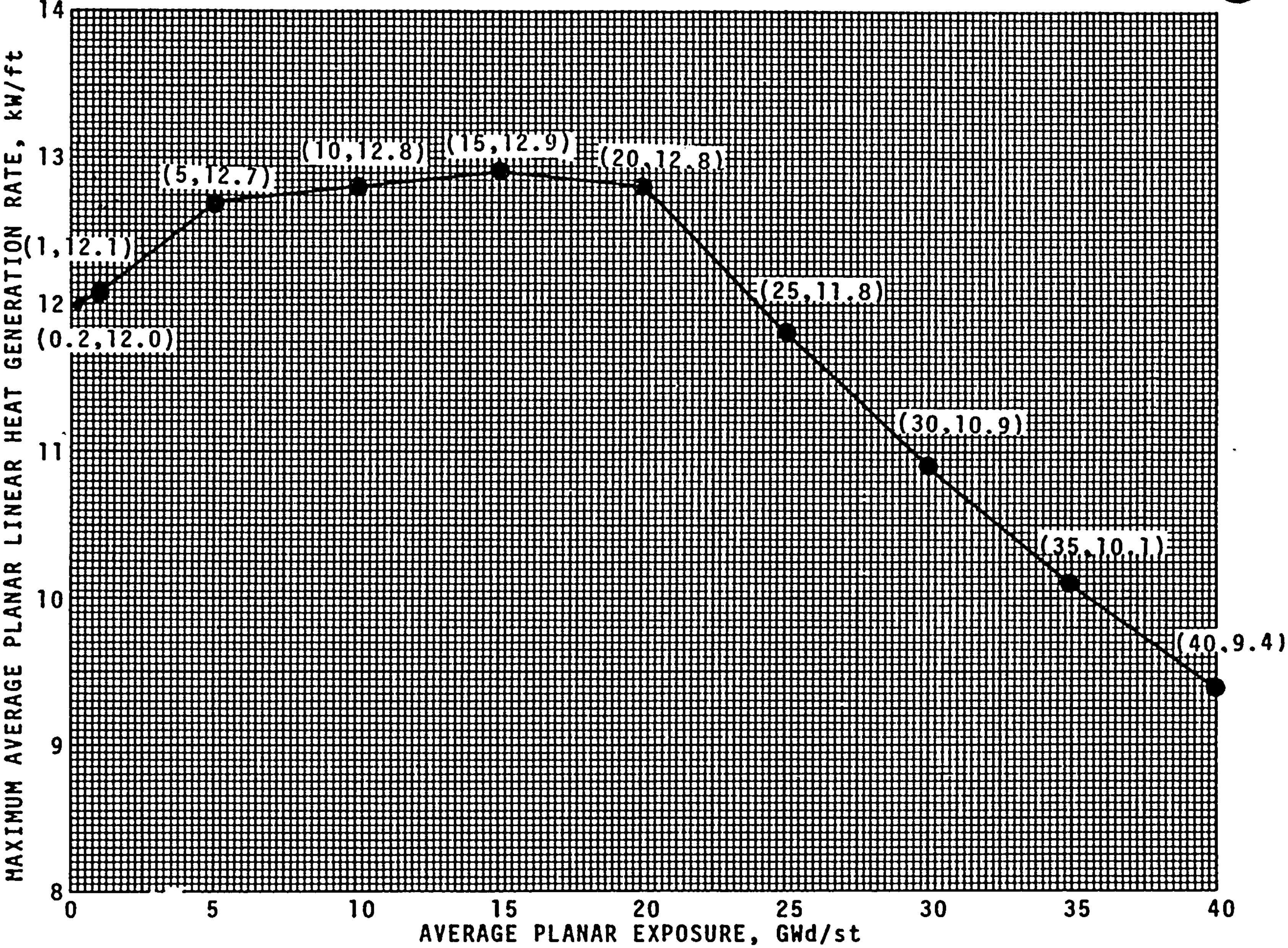

Figure 3.2.1-2 Maximum Average Planar Linear Heat Generation Rate vs. Average Planar Exposure, Fuel Type P8CRB176

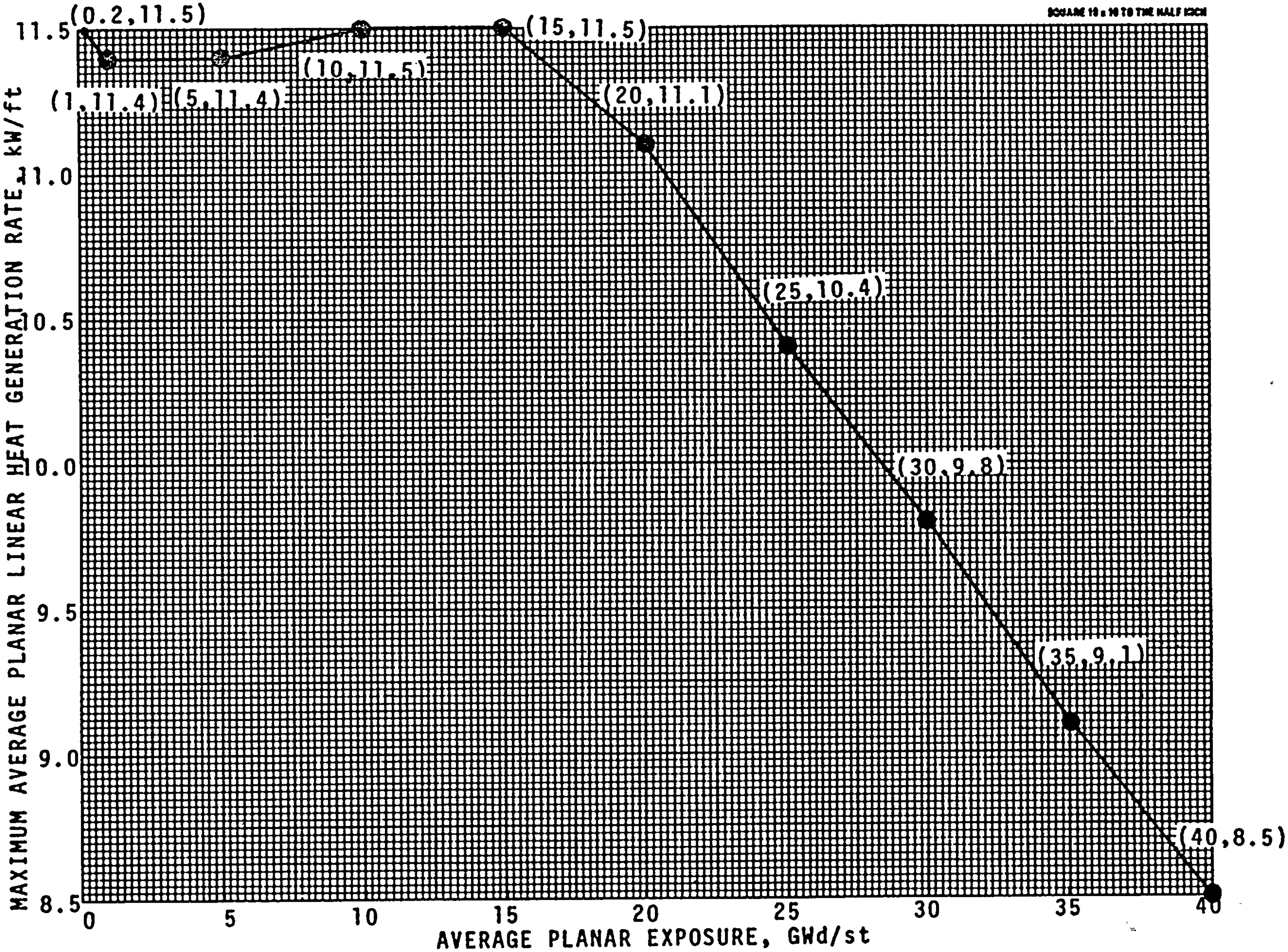

Figure 3.2.1-3 Maximum Average Planar Linear Heat Generation Rate vs. Average Planar Exposure, Fuel Type P8CRB071

POWER DISTRIBUTION LIMITS

3/4.2.2 AVERAGE POWER RANGE MONITOR SETPOINTS

LIMITING CONDITIONS FOR OPERATION

3.2.2 The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale 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†
$S \leq (0.66\ (W-\Delta W) + 51\%)T$	$S \leq (0.66\ (W-\Delta W) + 54\%)T$
$S_{RB} \leq (0.66\ (W-\Delta W) + 42\%)T$	$S_{RB} \leq (0.66\ (W-\Delta W) + 45\%)T$

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 108.5 million lb/hr.

T = The ratio FRACTION OF RATED THERMAL POWER divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY.

T is applied only if less than or equal to 1.0.

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

ACTION:

With the APRM flow-biased simulated thermal power-upscale 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.

---

* With CMFLPD greater than the FRTP rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that 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 adjustment is posted on the reactor control panel.

† The Average Power Range Monitor Scram and Rod Block Functions vary as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ΔW = 0 for two loop operation. ΔW = 5% for single loop operation.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (ODYN OPTION B)

LIMITING CONDITIONS FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit shown in Figure 3.2.3-1 times the K_f shown in Figure 3.2.3-2 with:

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

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.688 + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2} [0.052],$$

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

n = number of surveillance tests performed to date in cycle

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

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test

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

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

LIMITING CONDITIONS FOR OPERATION

3.2.3 (Continued)

ACTION:

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be equal to or greater than the MCPR limit shown in Figure 3.2.3-1 EOC-RPT inoperable curve times the K_f shown in Figure 3.2.3-2.

b. With the main turbine bypass system inoperable per Specification 3.7.7, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that, within 1 hour, MCPR is determined to be equal to or greater than the MCPR limit shown in Figure 3.2.3-1 main turbine bypass inoperable curve times the K_f shown in Figure 3.2.3-2.

c. With MCPR less than the applicable MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2, as applicable, initiate corrective action within 15 minutes to restore MCPR within the required limit. Restore MCPR to within the required limit within 4 hours, if necessary, by reducing THERMAL POWER to the level required.

SURVEILLANCE REQUIREMENTS

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

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

   1. At least once per 24 hours,
   2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
   3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR, or

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

   1. At least once per 24 hours,
   2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
   3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

c. The provisions of Specification 4.0.4 are not applicable.

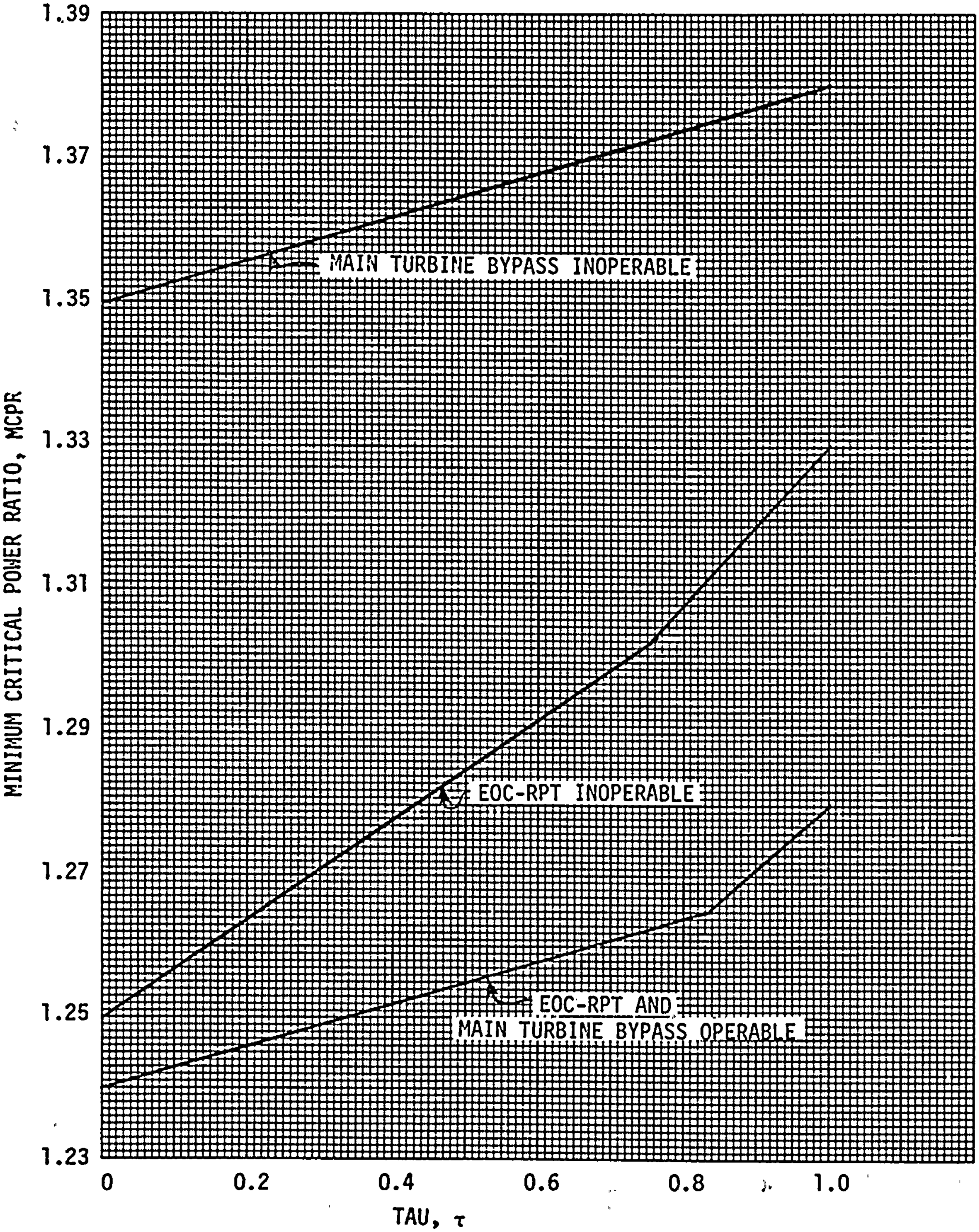

Figure 3.2.3-1 Minimum Critical Power Ratio vs. τ at Rated Flow

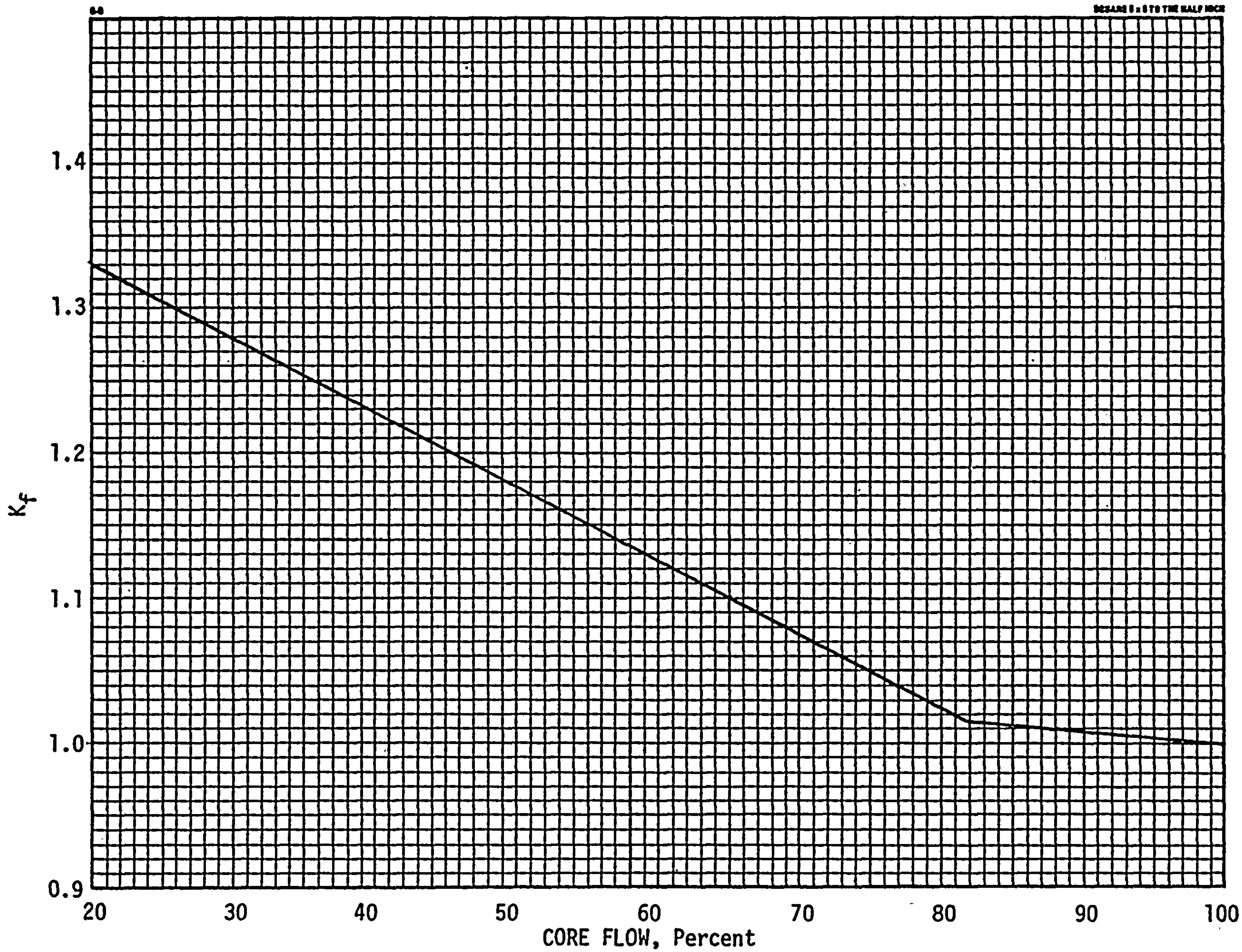

Figure 3.2.3-2 K_f as a Function of Percent Core Flow

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITIONS 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 LHGRs shall be determined to be equal to or less than the limit:

a. At least once per 24 hours,

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

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

d. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one Trip System, place the inoperable channel(s) and/or that Trip System in the tripped condition* within 1 hour. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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 3 3	1 2 3
b. Inoperative	2 3, 4 5	3 3 3	1 2 3
2. Average Power Range Monitor(c):			
a. Neutron Flux - Upscale, Setdown	2 3, 4 5(b)	2 2 2	1 2 3
b. Flow Biased Simulated Thermal Power - Upscale	1	2	4
c. Fixed Neutron Flux - Upscale	1	2	4
d. Inoperative	1, 2 3, 4 5	2 2 2	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

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

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) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, and the Refuel position one-rod-out interlock is OPERABLE per Specification 3.9.1, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

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

(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 PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

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

(i) This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 129.6** psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

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

** To allow for instrument accuracy, calibration and drift, a setpoint of less than or equal to 119 psig turbine first stage pressure shall be used.

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 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.

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

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

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than or equal to 129.6* psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

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

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

ACTION 9 - Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

* To allow for instrument accuracy, calibration, and drift, a setpoint of less than or equal to 119 psig turbine first-stage pressure shall be used.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (SECONDS)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤0.09**
c. Fixed Neutron Flux - Upscale	≤0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤0.55
4. Reactor Vessel Water Level - Low, Level 3	≤1.05
5. Main Steam Line Isolation Valve - Closure	≤0.06
6. Main Steam Line Radiation - High	NA
7. Drywell Pressure - High	NA
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter/Trip Unit	NA
b. Float Switch	NA
9. Turbine Stop Valve - Closure	≤0.06
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤0.08†
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA

* Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

** 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 FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U, S,(b) S	S/U(c), W, R(d) W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor(e):				
a. Neutron Flux - Upscale, Setdown	S/U, S,(b) S	S/U(c), W W	SA SA	2 3, 4, 5
b. Flow-Biased Simulated Thermal Power - Upscale	S, D(f)	S/U(c),W	W(g)(h), SA, R(i)	1
c. Fixed Neutron Flux - Upscale	S	S/U(c), W	W(g), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R(k)	1, 2
4. Reactor Vessel Water Level - Low, Level 3	S	M	R(k)	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2(j)
7. Drywell Pressure - High	S	M	R(k)	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[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Transmitter/Trip Unit	S	M	R(k)	1, 2, 5(m)
b. Float Switches	NA	M	R	1, 2, 5(m)
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	NA	M	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(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 before startup, if not performed within the previous 7 days.

(d) Perform a CHANNEL FUNCTIONAL TEST with the mode switch in Startup/Hot Standby and the plant in the COLD SHUTDOWN or REFUEL Condition.

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

(f) Verify measured core flow (total core flow) to be in the range of established core flow at the existing loop flow (APRM%).

(g) 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.

(h) This calibration shall consist of the adjustment of the APRM flow-biased channel to conform to a calibrated flow signal.

(i) This calibration shall consist of verifying the 6 ± 0.6 seconds simulated thermal power time constant.

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

(k) Perform the calibration procedure for the trip unit setpoint at least once per 31 days.

(l) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required to be OPERABLE per Special Test Exception 3.10.1.

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

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITIONS 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 shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

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

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one Trip System, place the inoperable channel(s) and/or that Trip System in the tripped condition* within 1 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 1 hour and take the ACTION required by Table 3.3.2-1.

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

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

INSTRUMENTATION

ISOLATION ACTUATION INSTRUMENTATION

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 so that all channels are tested at least once per 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(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. Primary Containment Isolation Signals				
a. Reactor Vessel Water Level				
1. Low, Low, Low, Level 1	1	2	1, 2, 3	20
2. Low, Low, Level 2(c)(d)	2,3,6,7,8,9	2	1, 2, 3 and *	20
3. Low, Level 3	4,5	2	1, 2, 3	20
b. Drywell Pressure - High(c)(d)	3,4,8,9	2	1, 2, 3	20
c. Main Steam Line				
1. Radiation - High(e)	1,2	2	1, 2, 3	21
2. Pressure - Low	1	2	1	23
3. Flow - High	1	2/Line	1, 2, 3	21
d. Main Steam Line Tunnel				
1. Temperature - High	1	2	1, 2, 3	21
2. ΔTemperature - High	1	2	1, 2, 3	21
3. Temperature - High MSL Lead Enclosure	1	6	1, 2, 3	21
e. Condenser Vacuum-Low	1	2	1, 2**, 3**	21
f. RHR Equipment Area Temperature - High (HXs/A&B Pump Rooms)	5,10	2	1, 2, 3	28
g. Reactor Vessel Pressure - High (RHR Cut-in Permissive)	5	2	1, 2, 3	28

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. Primary Containment Isolation Signals (Continued)				
h. SGTS Exhaust - High Radiation	9	1	1, 2, 3	27
i. RWCU System				
1) ΔFlow - High	6,7	1	1, 2, 3	22
2) ΔFlow - High, Timer	6,7	1	1, 2, 3	22
3) Standby Liquid Control, SLCS, Initiation	6(f),7(f)	1	1, 2, 5†	22
j. RWCU Equipment Area				
1) Pump Room A Temperature - High	6,7	1	1, 2, 3	22
2) Pump Room B Temperature - High	6,7	1	1, 2, 3	22
3) HX Room Temperature - High	6,7	1	1, 2, 3	22
k. Reactor Building Pipe Chase				
1) Azimuth 180° (Upper), Temperature - High	5,6,7,10	1	1, 2, 3	22
2) Azimuth 180° (Lower), Temperature - High	5,6,7,10	2	1, 2, 3	22
3) Azimuth 40°, Temperature - High	5,6,7,10	1	1, 2, 3	22
l. Reactor Building Temperature - High	5,10	5	1, 2, 3	22
m. Manual Isolation Pushbutton [NSSSS]	1	2	1, 2, 3	25
	2,4,5	2	1, 2, 3	26
	3,6,7	1	1, 2, 3	26
	8	2	1, 2, 3	25,27
	9	2	1, 2, 3	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL(a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(b)	APPLICABLE OPERATIONAL CONDITION	ACTION
2. RCIC Isolation Signals				
a. RCIC Steam Line Flow - High, Timer	10	1	1, 2, 3	22
b. RCIC Steam Supply Pressure - Low(g)	10, 11	2	1, 2, 3	22
c. RCIC Steam Line Flow - High	10	1	1, 2, 3	22
d. RCIC Turbine Exhaust Diaphragm Pressure - High(g)	10	2	1, 2, 3	22
e. RCIC Equipment Area Temperature - High	10	1	1, 2, 3	22
f. RCIC Steam Line Tunnel Temperature - High	10	1	1, 2, 3	22
g. Manual Isolation Push Button [RCIC](h)	10	1/Division I Only	1, 2, 3	26
h. Drywell Pressure - High(j)	11(i)	2	1, 2, 3	22
i. RHR/RCIC Steam Flow - High	10	1	1, 2, 3	22
3. Secondary Containment Isolation Signals				
a. Reactor Building Above the Refuel Floor Exhaust Radiation - High	(c)(d)	1	1, 2, 3 and ††	27
b. Reactor Building Below the Refuel Floor Exhaust Radiation - High	(c)(d)	1	1, 2, 3 and ††	27

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATIONS

* During CORE ALTERATIONS and operations with a potential for draining the reactor vessel. This applies to functions described in notes (c) and (d) that isolate secondary containment and automatically start the SGTS.

** When any turbine stop valve is greater than 90% open and/or when the key-locked condenser low vacuum bypass switch is open (in Normal position).

† Valves 2WCS*MOV102 and 2WCS*MOV112 are also required to be OPERABLE or closed in OPERATIONAL CONDITION 5 with any control rod withdrawn but not with control rods removed per Specifications 3.9.10.1 and 3.9.10.2.

†† When handling irradiated fuel in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) Refer to Table 3.6.3-1 for applicable valves in each valve group. Refer to Table 3.3.2-4 for valve groups, associated isolation signals and key to isolation signals.

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

(c) Also actuates the standby gas treatment system.

(d) Also actuates reactor building ventilation isolation dampers per Table 3.6.5.2-1.

(e) Also trips and isolates the air removal pumps.

(f) Initiation of SLCS pump 2SLS*P1B closes 2WCS*MOV102 and manual initiation of SLCS pump 2SLS*P1A closes 2WCS*MOV112.

(g) For this signal one Trip System has 2 channels which close valves 2ICS*MOV 128 and 2ICS*MOV 170, while the other Trip System has 2 channels which close 2ICS*MOV 121.

(h) Manual initiation only isolates 2ICS*MOV121 and only following manual or automatic initiation of the RCIC system.

(i) Only used in conjunction with low RCIC steam supply pressure and high drywell pressure to isolate 2ICS*MOV148 and 2ICS*MOV164.

(j) Signal from LPCS/RHR initiation circuitry.

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

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Not used.

ACTION 25 - Restore the manual isolation 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 26 - Restore the manual isolation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Establish REACTOR BUILDING INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 28 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Primary Containment Isolation Signals (Continued)		
a. Reactor Vessel Water Level*		
1) Low, Low, Low, Level 1	≥17.8 in.	≥10.8 in.
2) Low, Low, Level 2	≥108.8 in.	≥101.8 in.
3) Low, Level 3	≥159.3 in.	≥157.8 in.
b. Drywell Pressure - High	≤1.68 psig	≤1.88 psig
c. Main Steam Line		
1) Radiation - High	≤3x Full Power Background	≤3.6x Full Power Background
2) Pressure - Low	≥766 psig	≥746 psig
3) Flow - High	≤103 psid	≤109.5 psid
d. Main Steam Line Tunnel		
1) Temperature - High	≤159°F	≤169.5°F
2) ΔTemperature - High	≤50°F	≤62.8°F
3) Temperature - High MSL Lead Enclosure	≤140°F	≤150.5°F
e. Condenser Vacuum Low	≥8.5 in Hg vacuum	≥7.6 in. Hg vacuum
f. RHR Equipment Area Temperature - High (HXs/A&B Pump Rooms)	≤135°F	≤144.5°F
g. Reactor Vessel Pressure - High (RHR Cut-in Permissive)	≤128 psig	≤148 psig
h. SGTS Exhaust - High Radiation	≤5.7×10^{-3} μCi/cc	≤1.0×10^{-2} μCi/cc

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Primary Containment Isolation Signals (Continued)		
i. RWCU System		
1) Δ Flow - High	$\leq$150.5 gpm	$\leq$165.5 gpm
2) Δ Flow - High, Timer	$\leq$45 sec	$\leq$47 sec
3) Standby Liquid Control, SLCS, Initiation	NA	NA
j. RWCU Equipment Area Temperature		
1) Pump Room A Temperature - High	$\leq$135°F	$\leq$144.5°F
2) Pump Room B Temperature - High	$\leq$150°F	$\leq$159.5°F
3) HX Room Temperature - High	$\leq$135°F	$\leq$140.5°F
k. Reactor Building Pipe Chase		
1) Azimuth 180° (Upper), Temperature - High	$\leq$135°F	$\leq$144.5°F
2) Azimuth 180° (Lower), Temperature - High	$\leq$135°F	$\leq$140.5°F
3) Azimuth 40°, Temperature - High	$\leq$135°F	$\leq$140.5°F
l. Reactor Building Temperature - High	$\leq$130.2°F	$\leq$134°F
m. Manual Isolation Pushbutton [NSSSS]	NA	NA
2. RCIC Isolation Signals		
a. RCIC Steam Line Flow - High, Timer	$\geq$3 sec, <13 sec	13 sec
b. RCIC Steam Supply Pressure - Low	$\geq$75 psia	$\geq$70 psia
c. RCIC Steam Line Flow - High	$\leq$184.5 in. H_2O**	$\leq$193.0 in. H_2O**

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. RCIC Isolation Signals (Continued)		
d. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$10 psig	$\leq$20 psig
e. RCIC Equipment Area Temperature - High	$\leq$135°F	$\leq$140.5°F
f. RCIC Steam Line Tunnel Temperature - High	$\leq$135°F	$\leq$140.5°F
g. Manual Isolation Push Button [RCIC]	NA	NA
h. Drywell Pressure - High	$\leq$1.68 psig	$\leq$1.88 psig
i. RHR/RCIC Steam Flow - High	$\leq$96 in. H_2O**	$\leq$104.5 in. H_2O**
3. Secondary Containment Isolation Signals		
a. Reactor Building Above the Refuel Floor Exhaust Radiation - High	$\leq 2.36 \times 10^{-3}$ µCi/cc	$\leq 2.46 \times 10^{-3}$ µCi/cc
b. Reactor Building Below the Refuel Floor Exhaust Radiation - High	$\leq 2.36 \times 10^{-3}$ µCi/cc	$\leq 2.46 \times 10^{-3}$ µCi/cc

* See Bases Figure B3/4 3-1.

** Preliminary setpoint - actual setpoint to be determined during startup test program and submitted in writing within 90 days of their determination.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (SECONDS)
1. Primary Containment Isolation Signals	
a. Reactor Vessel Water Level	
1) Low, Low, Low, Level 1	$\leq$1.0
2) Low, Low, Level 2	NA
3) Low, Level 3	NA
b. Drywell Pressure - High	NA
c. Main Steam Line	
1) Radiation - High	NA
2) Pressure - Low	$\leq$1.0
3) Flow - High	$\leq$0.5
d. Main Steam Line Tunnel	
1) Temperature - High	NA
2) Δ Temperature - High	NA
3) Temperature - High MSL Lead Enclosure	NA
e. Condenser Vacuum - Low	NA
f. RHR Equipment Area Temperature - High (HXs/A&B Pump Rooms)	NA
g. Reactor Vessel Pressure - High (RHR Cut-in Permissive)	NA
h. SGTS Exhaust - High Radiation	NA
i. RWCU System	
1) Δ Flow - High	NA
2) Δ Flow - High, Timer	NA
3) Standby Liquid Control, SLCS, Initiation	NA
j. RWCU Equipment Area Temperature - High (HXs/A&B Pump Rooms)	NA
k. Reactor Building Pipe Chase Temperature - High	NA

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (SECONDS)
1. Primary Containment Isolation Signals (Continued)	
l. Reactor Building Temperature - High	NA
m. Manual Isolation Pushbutton [NSSSS]	NA
2. RCIC Isolation Signals	
a. RCIC Steam Line Flow - High, Timer	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Steam Line Flow - High	NA
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
e. RCIC Equipment Area Temperature - High	NA
f. RCIC Steam Line Tunnel Temperature - High	NA
g. Manual Isolation Push Button [RCIC]	NA
h. Drywell Pressure - High	NA
i. RHR/RCIC Steam Flow - High	NA
3. Secondary Containment Isolation Signals	
a. Reactor Building Above the Refuel Floor Exhaust Radiation - High	NA
b. Reactor Building Below the Refuel Floor Exhaust Radiation - High	NA

TABLE 3.3.2-4

VALVE GROUPS AND ASSOCIATED ISOLATION SIGNALS

VALVE GROUPS	ASSOCIATED CONTAINMENT ISOLATION VALVES BY FUNCTION	ISOLATION SIGNALS
1	MSIVs and MSL Drains	Z, X, C, D, E, P, T, R, RM, AA
2	Recirculation System Sample Valves	B, C, Z, RM
3	TIP Isolation	B, F, Z, RM
4	RHR Sample & Radioactive Waste Valves	A, Z, F, RM
5	RHR Shutdown Cooling Valves and Head Spray	A, L, M, Z, RM, CC, DD
6	RWCU Outboard Isolation Valve	B, U, J, S, Z, RM, DD
7	RWCU Inboard Isolation Valve	B, J, U, S, Z, RM, DD
8	All Containment Isolation Valves Not Assigned To Another Group	B, F, Z, RM
9	Containment Purge Valves	B, F, Y, Z, RM
10	RCIC Steam Supply Valves	K, M, H, Z, RM, BB, CC, DD
11	RCIC Vacuum Breaker Isolation Valves	H* & F*, RM
12	Remote Manually Operated Containment Valves	RM

* Both signals must be coincident to cause isolation.

TABLE 3.3.2-4 (Continued)

VALVE GROUPS AND ASSOCIATED ISOLATION SIGNALS

KEY TO ISOLATION SIGNALS

A	=	Low reactor vessel water, Level 3
B	=	Low reactor vessel water, Level 2
C	=	High main steam line radiation
D	=	High main steam line flow
E	=	High main steam line tunnel ambient temperature
F	=	High drywell pressure
G	=	Not used
H	=	Low RCIC steam supply pressure
J	=	High reactor water cleanup system equipment area ambient temperature
K	=	Reactor core isolation cooling pipe routing area high temperature and RCIC equipment area high temperature, high steam line flow, high turbine exhaust diaphragm pressure
L	=	High reactor vessel pressure
M	=	High residual heat removal system equipment area ambient temperatures
N	=	Not used
P	=	Low main steam line turbine inlet pressure
R	=	Low main condenser vacuum
S	=	Standby liquid control system actuated
T	=	High main steam line tunnel differential temperatures
U	=	High reactor water cleanup system differential flow
W	=	High reactor water cleanup system nonregenerative heat exchanger outlet temperature (not a containment isolation signal)
X	=	Low reactor water level, Level 1
Y	=	Standby gas treatment exhaust high radiation
LC	=	Locked closed

TABLE 3.3.2-4 (Continued)

VALVE GROUPS AND ASSOCIATED ISOLATION SIGNALS

KEY TO ISOLATION SIGNALS

RM	=	Remote manual switch from control room
LMC	=	Locked closed - position indicator
Z	=	Manual isolation
AA	=	Main steam line lead enclosure high ambient temperature
BB	=	RCIC/RHR steamline flow - high
CC	=	Reactor building high ambient temperature
DD	=	Reactor building pipe chase high ambient temperature

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTION TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Primary Containment Isolation Signals				
a. Reactor Vessel Water Level				
1) Low, Low, Low, Level 1	S	M	R(a)	1, 2, 3
2) Low, Low, Level 2	S	M	R(a)	1, 2, 3 and *
3) Low, Level 3	S	M	R(a)	1, 2, 3
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	M	R	1, 2, 3
2) Pressure - Low	S	M	R(a)	1
3) Flow - High	S	M	R(a)	1, 2, 3
d. Main Steam Line Tunnel				
1) Temperature - High	S	M	R(b)	1, 2, 3
2) ΔTemperature - High	S	M	R(b)	1, 2, 3
3) Temperature - High MSL Lead Enclosure	S	M	R(b)	1, 2, 3
e. Condenser Vacuum - Low	S	M	R(a)	1, 2**, 3**
f. RHR Equipment Area Temperature - High (HXs/A&B Pump Rooms)	S	M	R(b)	1, 2, 3
g. Reactor Vessel Pressure High (RHR Cut-in Permissive)	S	M	R(a)	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTION TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
1. Primary Containment Isolation Signals (Continued)				
h. SGTS Exhaust - High Radiation	NA	M	R	1, 2, 3
i. RWCU System				
1) ΔFlow - High	S	M	R	1, 2, 3
2) ΔFlow - High, Timer	NA	M	R	1, 2, 3
3) Standby Liquid Control, SLCS, Initiation	NA	R	NA	1, 2, 5††
j. RWCU Equipment Area				
1) Pump Room A Temperature - High	S	M	R(b)	1, 2, 3
2) Pump Room B Temperature - High	S	M	R(b)	1, 2, 3
3) HX Room Temperature - High	S	M	R(b)	1, 2, 3
k. Reactor Building Pipe Chase				
1) Azimuth 180° (Upper), Temperature - High	S	M	R(b)	1, 2, 3
2) Azimuth 180° (Lower), Temperature - High	S	M	R(b)	1, 2, 3
3) Azimuth 40°, Temperature - High	S	M	R(b)	1, 2, 3
l. Reactor Building Temperature - High	S	M	R(b)	1, 2, 3
m. Manual Isolation Pushbutton [NSSSS]	NA	M(c)	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTION TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
2. RCIC Isolation Signals				
a. RCIC Steam Line Flow - High, Timer	NA	M	R	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	M	R(a)	1, 2, 3
c. RCIC Steam Line Flow - High	S	M	R(a)	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R(a)	1, 2, 3
e. RCIC Equipment Area Temperature - High	S	M	R(b)	1, 2, 3
f. RCIC Steam Line Tunnel Temperature - High	S	M	R(b)	1, 2, 3
g. Manual Isolation Push Button [RCIC]	NA	M(c)	NA	1, 2, 3
h. Drywell Pressure - High	S	M	R(a)	1, 2, 3
i. RHR/RCIC Steam Flow - High	S	M	R(a)	1, 2, 3
3. Secondary Containment Isolation Signals				
a. Reactor Building Above the Refuel Floor Exhaust Radiation - High	NA	M	R	1, 2, 3, and †
b. Reactor Building Below the Refuel Floor Exhaust Radiation - High	NA	M	R	1, 2, 3, and †

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* During CORE ALTERATIONS and operations with a potential for draining the reactor vessel. This only applies to secondary containment isolation and automatic start of SGTS.

** When any turbine stop valve is greater than 90% open and/or when the key-locked condenser low vacuum bypass switch is open (in Normal position).

† When handling irradiated fuel in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

†† Valves 2WCS*MOV102 and 2WCS*MOV112 are required to be OPERABLE or closed in OPERATIONAL CONDITION 5 with any control rod withdrawn but not with control rods removed per Specifications 3.9.10.1 and 3.9.10.2.

(a) Perform the calibration procedure for the trip unit setpoint at least once per 31 days.

(b) Calibration excludes sensors; sensor response and comparison shall be done in lieu of.

(c) Manual isolation pushbuttons are tested at least once per operating cycle during shutdown. All other circuitry associated with manual isolation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of the circuitry required to be tested for the automatic system isolation.

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITIONS 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 Value 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, or
   2. 72 hours, provided either the HPCS or RCIC systems are inoperable.

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.3 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 so that all channels are tested at least once per 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/Pump	1, 2, 3, 4*, 5*	31
d. LPCS Injection Valve Permissive	1	1, 2, 3 4*, 5*	32 33
e. LPCI Injection Valve Permissive	1	1, 2, 3 4*, 5*	32 33
f. LPCI Pump A Start Time Delay Relay Normal Power	1	1, 2, 3, 4*, 5*	32
g. LPCI Pump A Start Time Delay Relay Emergency Power	1	1, 2, 3, 4*, 5*	32
h. LPCS Pump Start Time Delay Normal Power	1	1, 2, 3, 4*, 5*	32
i. LPCS Pump Start Time Delay Emergency Power	1	1, 2, 3, 4*, 5*	32
j. LPCI Pump A Discharge Flow - Low (Bypass)	1/Pump	1, 2, 3, 4*, 5*	31
k. Manual Initiation	1/Trip System	1, 2, 3, 4*, 5*	35
2. Automatic Depressurization System Trip System "A"(c)			
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	2(b)	1, 2, 3	30
b. ADS Timer	1	1, 2, 3	32
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
d. LPCS Pump Discharge Pressure - High (Permissive)	2	1, 2, 3	32
e. LPCI Pump A Discharge Pressure - High (Permissive)	2	1, 2, 3	32
f. Manual Inhibit	1	1, 2, 3	32
g. Manual Initiation	2/System	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 II Trip System			
1. RHR-B and 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. LPCI Injection Valve Permissive	1/Valve	1, 2, 3 4*, 5*	32 33
d. LPCI Pump B Start Time Delay Relay Normal Power	1	1, 2, 3, 4*, 5*	32
e. LPCI Pump C Start Time Delay Relay Normal Power	1	1, 2, 3, 4*, 5*	32
f. LPCI Pump B Start Time Delay Relay Emergency Power	1	1, 2, 3, 4*, 5*	32
g. LPCI Pump C Start Time Delay Relay Emergency Power	1	1, 2, 3, 4*, 5*	32
h. LPCI Pump Discharge Flow - Low (Bypass)	1/Pump	1, 2, 3, 4*, 5*	31
i. Manual Initiation	1/Trip System	1, 2, 3, 4*, 5*	35
2. Automatic Depressurization System Trip System "B"(c)			
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	2(b)	1, 2, 3	30
b. ADS Timer	1	1, 2, 3	32
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
d. LPCI Pump (B and C) Discharge Pressure - High (Permissive)	2/Pump	1, 2, 3	32
e. Manual Inhibit	1	1, 2, 3	32
f. Manual Initiation	2/System	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 III Trip System			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4(b)	1, 2, 3, 4*, 5*	36
b. Drywell Pressure - High (d)	4(b)	1, 2, 3	36
c. Reactor Vessel Water Level - High, Level 8	4(e)	1, 2, 3, 4*, 5*	32
d. Condensate Storage Tank Level - Low	2(f)	1, 2, 3, 4*, 5*	37
e. Suppression Pool Water Level - High	2(f)	1, 2, 3, 4*, 5*	37
f. HPCS System Flow Rate - Low (Bypass)	1	1, 2, 3, 4*, 5*	31
g. Pump Discharge Pressure - High (Bypass)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation (d)	1/System	1, 2, 3, 4*, 5*	35

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. Loss of Power (Divisions I & II)					
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39
E. Loss of Power, Division III					
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TABLE NOTATIONS

* 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.

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of 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 associated division diesel generator.

(c) Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

(d) The injection function of Drywell Pressure High and Manual Initiation is not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than level 8 setpoint coincident with the vessel pressure less than 600 psig because of hot calibration/cold operation level error.

(e) Provides signal to close HPCS pump injection valve only.

(f) Provides signal to HPCS pump suction valves only.

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 1 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 1 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.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 1 hour.

ACTION 34 - Not Used.

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 1 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 1 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.

ACTION 39 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour*; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

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* The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. **Division I Trip System**		
1. **RHR-A (LPCI Mode) & LPCS System**		
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	≥17.8 in.*	≥10.8 in.
b. Drywell Pressure - High	≤1.68 psig	≤1.88 psig
c. LPCS Pump Discharge Flow - Low (Bypass)	≥1200 gpm	≥1000 gpm
d. LPCS Injection Valve Permissive	≤88 psid, decreasing	≤98 psid, decreasing
e. LPCI Injection Valve Permissive	≤130 psid, decreasing	≤150 psid, decreasing
f. LPCI Pump A Start Time Delay Relay Normal Power	≤5 sec	≤7 sec
g. LPCI Pump A Start Time Delay Relay Emergency Power	≤1 sec	≤2 sec
h. LPCS Pump Start Time Delay Normal Power	≤10 sec	≤12 sec
i. LPCS Pump Start Time Delay Emergency Power	≤6 sec	≤6.75 sec
j. LPCI Pump A Discharge Flow - Low (Bypass)	>1400 gpm	>1200 gpm
k. Manual Initiation	NA	NA
2. **Automatic Depressurization System Trip System "A"**		
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	≥17.8 in.*	≥10.8 in.
b. ADS Timer	<105 sec	<117 sec
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	≥159.3 in.*	≥157.8 in.
d. LPCS Pump Discharge Pressure - High (Permissive)	≥145 psig, increasing	≥125 psig, increasing
e. LPCI Pump A Discharge Pressure - High (Permissive)	≥125 psig, increasing	≥115 psig, increasing
f. Manual Inhibit	NA	NA
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 II Trip System		
1. RHR-B and C (LPCI Mode)		
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	$\geq$17.8 in.*	$\geq$10.8 in.
b. Drywell Pressure - High	$\leq$1.68 psig	$\leq$1.88 psig
c. LPCI Injection Valve Permissive	$\leq$130 psid, decreasing	$\leq$150 psid, decreasing
d. LPCI Pump B Start Time Delay Relay Normal Power	$\leq$5 sec	$\leq$7 sec
e. LPCI Pump C Start Time Delay Relay Normal Power	$\leq$10 sec	$\leq$12 sec
f. LPCI Pump B Start Time Delay Relay Emergency Power	$\leq$1 sec	$\leq$2 sec
g. LPCI Pump C Start Time Delay Emergency Power	$\leq$6 sec	$\leq$7 sec
h. LPCI Pump Discharge Flow - Low (Bypass)	$\geq$1400 gpm	$\geq$1200 gpm
i. Manual Initiation	NA	NA
2. Automatic Depressurization System Trip System "B"		
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	$\geq$17.8 in.*	$\geq$10.8 in.
b. ADS Timer	$\leq$105 sec	$\leq$117 sec
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	$\geq$159.3 in.*	$\geq$157.8 in.
d. LPCI Pump (B and C) Discharge Pressure - High (Permissive)	$\geq$125 psig, increasing	$\geq$115 psig, increasing
e. Manual Inhibit	NA	NA
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 III Trip System		
1. HPCS System		
a. Reactor Vessel Water Level - Low, Low, Level 2	≥108.8 in.*	≥101.8 in.
b. Drywell Pressure - High	≤1.68 psig	≤1.88 psig
c. Reactor Vessel Water Level - High, Level 8	≤202.3 in.*	≤209.3 in.
d. Condensate Storage Tank Level - Low	≥12.5 ft.**	≥12.25 ft.**
e. Suppression Pool Water Level - High	≤201.0 ft. el	≤201.1 ft. el
f. HPCS System Flow Rate - Low (Bypass)	≥825 gpm	≥750 gpm
g. Pump Discharge Pressure - High (Bypass)	≥240 psig	≥220 psig
h. Manual Initiation	NA	NA
D. Loss of Power (Divisions I & II)		
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	a. 4.16-kV basis - ≥3148	≥3051 volts
	b. ≤3.06-sec time delay	≤3.12-sec time delay
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	a. 4.16-kV basis - ≥3847 volts	≥3770 volts
	b. ≤8.16-sec time delay†	≤8.32-sec time delay†
	c. ≤30.6-sec time delay	≤31.2-sec time delay

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
E. Loss of Power (Division III)		
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	a. 4.16-kV basis - ≥3148 volts	≥3051 volts
	b. ≤3.06-sec time delay	≤3.12-sec time delay
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	a. 4.16-kV basis - ≥3847 volts	≥3770 volts
	b. ≤12.24-sec time delay	≤12.48-sec time delay

* See Bases Figure B3/4 3-1.

** Reference zero point for the CST is the bottom of the CST (el. 251 ft 0 in.)

† Alarm only without LOCA signal present; Alarm and trip with LOCA signal present.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (SECONDS)
1. Low-Pressure Core Spray System	
a. Diesel Generator Start Time*	≤10
b. Injection Valve Opening Time**	≤20
2. Low-Pressure Coolant Injection Mode of RHR System (Division I and Division II)	
a. Diesel Generator Start Time*	≤10
b. Injection Valve Opening Time**	≤20
3. Automatic Depressurization System	NA
4. High-Pressure Core Spray System	≤27
5. Loss of Power	NA

*Time to rated speed and voltage upon receipt of start signal.

**Time from receipt of open signal to full open.

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 IS 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(c)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(c)	1, 2, 3
c. LPCS Pump Discharge Flow - Low (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
d. LPCS Injection Valve Permissive	S	M	R(c)	1, 2, 3, 4*, 5*
e. LPCI Injection Valve Permissive	S	M	R(c)	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay Normal Power	NA	M	R	1, 2, 3, 4*, 5*
g. LPCI Pump A Start Time Delay Relay Emergency Power	NA	M	R	1, 2, 3, 4*, 5*
h. LPCS Pump Start Time Delay Normal Power	NA	M	R	1, 2, 3, 4*, 5*
i. LPCS Pump Start Time Delay Emergency Power	NA	M	R	1, 2, 3, 4*, 5*
j. LPCI Pump A Discharge Flow - Low (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
k. Manual Initiation	NA	M(a)	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(c)	1, 2, 3
b. ADS Timer	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	S	M	R(c)	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 IS REQUIRED
A. Division I Trip System (Continued)				
2. Automatic Depressurization System Trip System "A"** (Continued)				
d. LPCS Pump Discharge Pressure - High (Permissive)	S	M	R(c)	1, 2, 3
e. LPCI Pump A Discharge Pressure - High (Permissive)	S	M	R(c)	1, 2, 3
f. Manual Inhibit	NA	M	NA	1, 2, 3
g. Manual Initiation	NA	M(a)	NA	1, 2, 3
B. Division II Trip System				
1. RHR-B and C (LPCI Mode)				
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	S	M	R(c)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(c)	1, 2, 3
c. LPCI Injection Valve Permissive	S	M	R(c)	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay Normal Power	NA	M	R	1, 2, 3, 4*, 5*
e. LPCI Pump C Start Time Delay Relay Normal Power	NA	M	R	1, 2, 3, 4*, 5*
f. LPCI Pump B Start Time Delay Emergency Power	NA	M	R	1, 2, 3, 4*, 5*
g. LPCI Pump C Start Time Delay Relay Emergency Power	NA	M	R	1, 2, 3, 4*, 5*
h. LPCI Pump Discharge Flow - Low (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
i. Manual Initiation	NA	M(a)	NA	1, 2, 3, 4*, 5*

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 IS REQUIRED
B. Division II Trip System (Continued)				
2. Automatic Depressurization System Trip System "B"** (Continued)				
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	S	M	R(c)	1, 2, 3
b. ADS Timer	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	S	M	R(c)	1, 2, 3
d. LPCI Pump (B and C) Discharge Pressure - High (Permissive)	S	M	R(c)	1, 2, 3
e. Manual Inhibit	NA	M	NA	1, 2, 3
f. Manual Initiation	NA	M(a)	NA	1, 2, 3
C. Division III Trip System				
1. HPCS System				
a. Reactor Vessel Water Level - Low, Low, Level 2	S	M	R(c)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High(b)	S	M	R(c)	1, 2, 3
c. Reactor Vessel Water Level - High, Level 8	S	M	R(c)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R(c)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(c)	1, 2, 3, 4*, 5*
f. HPCS System Flow Rate - Low (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
g. Pump Discharge Pressure - High (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
h. Manual Initiation(b)	NA	M(a)	NA	1, 2, 3, 4*, 5*

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 IS REQUIRED
D. Loss of Power (Divisions I & II)				
1. 4.16-kV Emergency Bus Undervoltage - Loss of Voltage	S	M	R	1, 2, 3, 4†, 5†
2. 4.16-kV Emergency Bus Undervoltage - Degraded Voltage	S	M	R	1, 2, 3, 4†, 5†
E. Loss of Power (Division III)				
1. 4.16-kV Emergency Bus Undervoltage - Loss of Voltage	S	M	R	1, 2, 3, 4†, 5†
2. 4.16-kV Emergency Bus Undervoltage - Degraded Voltage	S	M	R	1, 2, 3, 4†, 5†

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM

ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

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

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

† Required when ESF equipment is required to be OPERABLE.

(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 actuation.

(b) The injection function of Drywell Pressure - High and Manual Initiation is not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the vessel pressure less than 600 psig due to the hot calibration/cold operation level error.

(c) Perform the calibration procedure for the Trip Setpoint at least once per 31 days.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITIONS 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 Value 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, place the inoperable channel(s) in the tripped condition within 1 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 reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition* within 1 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 1 hour or be in at least STARTUP within the next 6 hours.

---

* The inoperable channels need not be placed in the tripped condition if this would cause the Trip Function to occur. In this case, the inoperable channels shall be restored to OPERABLE status within 2 hours, or the Trip System shall be declared inoperable.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

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

* One Trip System may be placed in an inoperable status for up to 2 hours for required surveillance provided the other Trip System 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	≥108.8 in.*	≥101.8 in.
2. Reactor Vessel Pressure - High	≤1050 psig	≤1065 psig

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* 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*

* Perform the calibration procedure for the trip unit setpoint at least once per 31 days.

INSTRUMENTATION

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

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

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

ACTION:

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

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both Trip Systems, place the inoperable channel(s) in the tripped condition within 1 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 1 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 take the ACTION required by Specification 3.2.3.

e. With both Trip Systems inoperable, restore at least one Trip System to OPERABLE status within 1 hour or take the ACTION required by Specification 3.2.3.

INSTRUMENTATION

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

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, so that both types of channel inputs are tested at least once per 36 months.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve - Fast Closure	2**

* 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.

** This function shall be automatically bypassed when turbine first-stage pressure is less than or equal to 129.6 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER. To allow for instrument accuracy, calibration, and drift, a setpoint of less than or equal to 119 psig shall be used.

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

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$190
2. Turbine Control Valve - Fast Closure	$\leq$190

TABLE 4.3.4.2-1

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

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITIONS 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 an 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
1. Reactor Vessel Water Level - Low, Low, Level 2	2	50
2. Reactor Vessel Water Level - High, Level 8(b)	2	50
3. Condensate Storage Tank "A" Water Level - Low	2(c)	51
4. Manual Initiation(d)	1/system(e)	52

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 other OPERABLE channel in the same Trip System is monitoring that parameter.

(b) The RCIC Level 8 trip may be bypassed to perform RCIC 150 psig operational surveillance test in accordance with Specification 4.7.4.c.2.

(c) One Trip System with one-out-of-two logic.

(d) Manual initiation is not required to be OPERABLE with indicated reactor vessel water level on the wide-range instrument greater than the Level 8 setpoint coincident with the vessel pressure less than 600 psig due to the hot calibration/cold operation level error.

(e) One Trip System with one channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

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

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

b. For both Trip Systems with more than one channel inoperable, declare the RCIC system inoperable.

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

ACTION 52 - With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Low, Level 2	$\geq$108.8 in.*	$\geq$101.8 in.
2. Reactor Vessel Water Level - High, Level 8	$\leq$202.3 in.*	$\leq$209.3 in.
3. Condensate Storage Tank Level "A" - Low	$\geq$6.15 ft**	$\geq$5.9 ft**
4. Manual Initiation	NA	NA

* See Bases Figure B3/4 3-1.

** Reference zero point for the CST is the bottom of the CST (El. 251 ft 0 in.).

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low, Low, Level 2	S	M	R*
2. Reactor Vessel Water Level - High, Level 8	S	M	R*
3. Condensate Storage Tank Level - Low	S	M	R*
4. Manual Initiation**	NA	M†	NA

---

* Perform the calibration procedure for the trip unit setpoint at least once per 31 days.

** Manual initiation is not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the vessel pressure less than 600 psig because of the hot calibration/cold operation level error.

† 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 actuation.

INSTRUMENTATION

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Rod Block Monitor(a)			
a. Upscale	2	1*	60
b. Inoperative	2	1*	60
c. Downscale	2	1*	60
2. APRM			
a. Flow-Biased Neutron Flux - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. Source Range Monitor			
a. Detector Not Full In(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. Intermediate Range Monitor			
a. Detector Not Full In	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
5. Scram Discharge Volume Water Level - High, Float Switch	2	1, 2, 5**	62
6. Reactor Coolant System Recirculation Flow			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
7. Reactor Mode Switch			
a. Shutdown Mode	2	3, 4	62
b. Refuel Mode	2	5	62

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATIONS

* With THERMAL POWER greater than or equal to 30% of RATED THERMAL POWER.

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

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

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

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

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

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

ACTION

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

ACTION 61 - With the number of OPERABLE Channels:

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

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

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within 1 hour.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Rod Block Monitor		
a. Upscale	≤0.66 (W-ΔW)* + 40%	≤0.66 (W-ΔW)* + 43%
b. Inoperative	NA	NA
c. Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
2. APRM		
a. Flow-Biased Neutron Flux - Upscale	≤0.66 (W-ΔW)* + 42%	≤0.66 (W-ΔW)* + 45%
b. Inoperative	NA	NA
c. Downscale	≥4% 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 Monitor		
a. Detector Not Full In	NA	NA
b. Upscale	≤1 x 10^5 cps	≤1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥3 cps**	≥1.8 cps**
4. Intermediate Range Monitors		
a. Detector not full in	NA	NA
b. Upscale	≤108/125 divisions of full scale	≤110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥5/125 divisions of full scale	≥3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
Water Level - High, Float Switch	≤16.5 in.	≤39.75 in.

TABLE 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. Reactor Coolant System Recirculation Flow		
a. Upscale	<108% rated flow	<111% rated flow
b. Inoperative	NA	NA
c. Comparator	≤10% flow deviation	≤11% flow deviation
7. Reactor Mode Switch		
a. Shutdown Mode	NA	NA
b. Refuel Mode	NA	NA

---

* The rod block function is varied as a function of recirculation loop flow (W), and must be maintained in accordance with note (a) of Table 2.2.1-1. The trip setting of this average power range monitor function must also be maintained in accordance with Specification 3.2.2 and note (a) of Table 2.2.1-1.

** For initial loading and startup the count rate may be less than 3 cps if the following conditions are met: the signal to noise ratio is greater than or equal to 20, and the signal is greater than 0.7 cps.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Rod Block Monitor				
a. Upscale	NA	S/U(b)(c), M(c)	Q	1*
b. Inoperative	NA	S/U(b)(c), M(c)	NA	1*
c. Downscale	NA	S/U(b)(c), M(c)	Q	1*
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

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
5. Scram Discharge Volume				
Water Level - High, Float Switch	NA	M	R	1, 2, 5**
6. Reactor Coolant System Recirculation Flow				
a. Upscale	NA	S/U(b), M	Q	1
b. Inoperative	NA	S/U(b), M	NA	1
c. Comparator	NA	S/U(b), M	Q	1
7. Reactor Mode Switch				
a. Shutdown Mode	NA	R	NA	3, 4
b. Refuel Mode	NA	R	NA	5

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* With THERMAL POWER greater than or equal to 30% of RATED THERMAL POWER.

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

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

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

(c) Includes reactor manual control multiplexing system input.

INSTRUMENTATION

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITIONS 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, SOURCE 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 (a)	ACTION
1. Main Control Room Ventilation Radiation Monitors	2/System(b)	1, 2, 3, 5, and *	$\leq 5.92 \times 10^{-6}$ µCi/cc(c)	74
2. Area Monitors				
a. Criticality Monitor (New Fuel Storage Vault)	1	**	$\leq 1.0 \times 10^{2}$ mR/hr(d)	76
b. Control Room Direct Radiation Monitor	1	At all times	$\leq 2.5 \times 10^{-1}$ mR/hr(d)	76

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

TABLE NOTATIONS

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

** With fuel in the new fuel storage vault.

(a) Above measured background.

(b) Two Trip Systems, one for each special filter train and associated bypass valve, are provided with two channels per Trip System.

(c) Initiates control room emergency filtration with both channels of one Trip System at high setpoint.

(d) Alarm only.

ACTION

ACTION 72 - Deleted.

ACTION 73 - Deleted.

ACTION 74 -
a. With the number of OPERABLE channels in one or both Trip Systems one less than the minimum number of OPERABLE channels required, place the inoperable channel in the tripped condition within 8 hours.

b. With the number of OPERABLE channels in one Trip System two less than the minimum number of OPERABLE channels required, restore at least one of the inoperable channels to OPERABLE status within 7 days, or within the next 6 hours ensure operation of the control room emergency filtration system in the filtration mode of operation.

c. With the number of OPERABLE channels in both Trip Systems two less than the minimum of OPERABLE channels required, within 1 hour, ensure operation of the control room emergency filtration system in the filtration mode of operation.

ACTION 75 - Deleted.

ACTION 76 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once every 24 hours.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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

** With fuel in the new fuel storage vault.

SEISMIC MONITORING INSTRUMENTATION

LIMITING CONDITIONS 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 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelerographs and Trigger		
a. Reactor Bldg. Mat. El. 175'-0"	0 ± 1.0 g	1
b. Reactor Bldg. Refueling Floor El. 353'-10"	0 ± 1.0 g	1
c. Control Bldg. Mat. El. 214'-0"	0 ± 1.0 g	1
2. Triaxial Peak Accelerographs		
a. Diesel Generator Bldg. Service Water Piping	0-5 g*	1
b. Prim. Cont. High Pr. Core Spray Piping	0-10 g*	1
c. Prim. Cont. Recirc. Pump Motor	0-10 g*	1
3. Triaxial Seismic Switches		
Reactor Bldg. Mat. El. 175'-0"	0.025-0.25 g (Adjustable)	1**
4. Triaxial Response-Spectrum Recorders		
a. Reactor Bldg. Mat. El. 175'-0"	0 ± 2 g*	1**
b. Prim. Cont. RHR Piping Pene. El. 294'-6"	0 ± 2 g*	1
c. Reactor Bldg. Refueling Fl. El. 353'-10"	0 ± 2 g*	1
d. Control Bldg. Mat. El. 214'-0"	0 ± 2 g*	1

* Calibration required to be for the range: ± 1g.

** With control room annunciation.

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 and Trigger			
a. Reactor Bldg. Mat. El. 175'-0"	M*	SA	R
b. Reactor Bldg. Refueling Fl. El. 353'-10"	M*	SA	R
c. Control Bldg. Mat. El. 214'-0"	M*	SA	R
2. Triaxial Peak Accelerographs			
a. Diesel Gen. Bldg. Service Water Piping	NA	NA	R**
b. Prim. Cont. High Pr. Core Spray Piping	NA	NA	R**
c. Prim. Cont. Recirc. Pump Motor	NA	NA	R**
3. Triaxial Seismic Switches			
Reactor Bldg. Mat. El. 175'-0"	M***	SA	R
4. Triaxial Response-Spectrum Recorders			
a. Reactor Bldg. Mat. El. 175'-0"	M	SA	R**
b. Prim. Cont. RHR Piping Pene. El. 294'-6"	NA	NA	R**
c. Reactor Bldg. Refueling Fl. El. 353'-10"	NA	NA	R**
d. Control Bldg. Mat. El. 214'-0"	NA	NA	R**

* Battery and Trigger only.

** Calibration required to be for the range: ± 1 g.

*** Except seismic trigger.

INSTRUMENTATION

MONITORING INSTRUMENTATION

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT		ELEVATION	MINIMUM INSTRUMENTS OPERABLE
1.	Wind Speed	30 ft	1
		200 ft	1
2.	Wind Direction	30 ft	1
		200 ft	1
3.	Air Temperature Difference	30/200 ft	1

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

MONITORING INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

---

* Includes transfer switches associated with remote shutdown system controls.

TABLE 3.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Service Water Pump Disch Flow	2CES*PNL405	2/Division
2. Reactor Vessel Pressure	2CES*PNL405	1/Division
3. RX Vessel Water Level Wide Range	2CES*PNL405	1/Division
4. RX Vessel Water Level Narrow Range	2CES*PNL405	1/Division
5. RCIC Turbine Speed	2CES*PNL405	1
6. Suppression Pool Water Level	2CES*PNL405	1/Division
7. RHR Loop "A" Flow	2CES*PNL405	1
8. RHR Ht. Ex. Service Water "A" Outlet Flow	2CES*PNL405	1
9. Suppression Pool Temperature	2CES*PNL405	1/Division
10. RHR Loop "B" Flow	2CES*PNL405	1
11. RHR Ht. Ex. Service Water "B" Outlet Flow	2CES*PNL405	1
12. Safety/Relief Valve Position	2CES*PNL405	1/Valve
13. RCIC Flow Indicator/Controller	2CES*PNL405	1

TABLE 3.3.7.4-2

REMOTE SHUTDOWN SYSTEM CONTROLS

SYSTEM/SUBSYSTEM*	SYSTEMS/SUBSYSTEMS	MINIMUM OPERABLE SYSTEMS/ SUBSYSTEMS
1. RCIC System	1	1
2. RHR System		
A. Shutdown Cooling Mode	2	1/Division
B. Suppression Pool Cooling Mode	2	1/Division
3. Service Water System		
A. Pumps	6	2/Division
B. Supply Valves to Division I Division II Diesels	1/Division	1/Division
4. ADS System (Pressure Relief)	4 Valves/Division	4 Valves/Division
5. Nuclear Steam Supply Shutoff System (Isolation Groups 4 & 5 Reset)	1/Division	1/Division
6. Nitrogen Supply to ADS Accumulator Tanks	1/Division	1/Division

* Includes applicable transfer switches

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CALIBRATION	READOUT LOCATION
1. Service Water Pump Discharge Flow	M	R	2CES*PNL405
2. Reactor Vessel Pressure	M	R	2CES*PNL405
3. RX Vessel Water Level Wide Range	M	R	2CES*PNL405
4. RX Vessel Water Level Narrow Range	M	R	2CES*PNL405
5. RCIC Turbine Speed	R	R	2CES*PNL405
6. Suppression Pool Water Level	M	R	2CES*PNL405
7. RHR Loop "A" Flow	M	R	2CES*PNL405
8. RHR Ht. Ex. Service Water "A" Flow	M	R	2CES*PNL405
9. Suppression Pool Temp.	M	R**	2CES*PNL405
10. RHR Loop "B" Flow	M	R	2CES*PNL405
11. RHR Ht. Ex. Service Water "B" Outlet Flow	M	R	2CES*PNL405
12. Safety/Relief Valve Position (4 Valves)	M	R*	2CES*PNL405
13. RCIC Flow Indicator/Controller	R	R	2CES*PNL405

* CHANNEL calibration is performed per Specification 4.4.2

** CHANNEL calibration excludes sensors; sensor comparison shall be done in lieu of sensor calibration.

INSTRUMENTATION

MONITORING INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	80
2. Reactor Vessel Water Level				
a. Fuel Zone	2	1	1, 2, 3	80
b. Wide Range	2	1	1, 2, 3	80
3. Suppression Pool Water Level				
a. Narrow Range	2	1	1, 2, 3	83
b. Wide Range	2	1	1, 2, 3	83
4. Suppression Pool Water Temperature	8, 2/Quadrant	4, 1/Quadrant	1, 2	80
5. Suppression Chamber Pressure	2	1	1, 2	80
6. Suppression Chamber Air Temperature	2	1	1, 2	80
7. Drywell Pressure				
a. Narrow Range	2	1	1, 2	80
b. Wide Range	2	1	1, 2	80
8. Drywell Air Temperature	2	1	1, 2	80
9. Drywell Oxygen Concentration	2	1	1, 2	80
10. Drywell Hydrogen Concentration Analyzer and Monitor	2	1	1, 2	80
11. Safety/Relief Valve Position Indicators*	2/Valve	1/Valve	1, 2	80

TABLE 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
12. Drywell High Range Radiation Monitors	2	1	1, 2, 3	85
13. RHR Heat Exchanger Service Water Radiation Monitor	1/Heat Exchanger	1/Heat Exchanger	1, 2, 3	81
14. Refuel Platform Area Radiation Monitor	1	1	**	82
15. Neutron Flux†				
APRM	2	1	1, 2	80
IRM	2	1	1, 2	80
SRM	2	1	1	80
16. Primary Containment Isolation Valve Position Indication	1	1	1, 2	84

*Acoustic monitoring and tail pipe temperature
**When handling fuel, or components in the fuel pool or reactor cavity.
†Neutron flux indication is sufficient to meet the Operability requirement of this specification.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 80 - a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

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

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

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

b. In lieu of another report required by Specification 6.9.2, 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.

ACTION 82 - With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, suspend movement of fuel or components in the fuel pool or reactor cavity, or, initiate the preplanned alternate method of monitoring the appropriate parameter(s).

ACTION 83 - 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 and in COLD SHUTDOWN within the following 24 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirement 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 and in COLD SHUTDOWN within the following 24 hours.

Action 84 - Take the action commensurate with Specification 3.6.3.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 85 - 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 channels to OPERABLE status within 7 days or initiate the preplanned alternate method of monitoring the appropriate parameter.

With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

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

b. In lieu of another report required by Specification 6.9.2, 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			
a. Fuel Zone	M	R	1, 2, 3
b. Wide Range	M	R	1, 2, 3
3. Suppression Pool Water Level			
a. Narrow Range	M	R	1, 2, 3
b. Wide Range	M	R	1, 2, 3
4. Suppression Pool Water Temperature	M	R*	1, 2
5. Suppression Chamber Pressure	M	R	1, 2
6. Suppression Chamber Air Temperature	M	R*	1, 2
7. Drywell Pressure			
a. Narrow Range	M	R	1, 2
b. Wide Range	M	R	1, 2
8. Drywell Air Temperature	M	R*	1, 2
9. Drywell Oxygen Concentration	M	R	1, 2
10. Drywell Hydrogen Concentration Analyzer and Monitor	M	Q**	1, 2
11. Safety/Relief Valve Position Indicators	M	R	1, 2
12. Drywell High Range Radiation Monitors	M	R†	1, 2, 3
13. RHR Heat Exchanger Service Water Radiation Monitor	M	R	1, 2, 3
14. Refuel Platform Area Radiation Monitor	M	R	††
15. Neutron Flux			
a. APRM	M	R	1, 2
b. IRM	M	R	1, 2
c. SRM	M	R	1
16. Primary Containment Isolation Valve Position Indication	M†††	R***	1, 2

TABLE 4.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* Excludes sensors; sensor comparison shall be done in lieu of sensor calibration.

** Using sample gas containing:

a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

***The CHANNEL CALIBRATION shall consist of position indication verification using ASME Section XI IWV-3300 test criteria.

† 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.

†† When handling fuel or components in the fuel pool or reactor cavity.

†††Red, Green or other indication shall be verified as indicating valve position.

SOURCE RANGE MONITORS

LIMITING CONDITIONS FOR OPERATION

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

a. In OPERATIONAL CONDITION 2*, three
b. In OPERATIONAL CONDITIONS 3 and 4, two.

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

ACTION:

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

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

---

* With IRMs on range 2 or below.

** Neutron detectors may be excluded from CHANNEL CALIBRATION.

INSTRUMENTATION

MONITORING INSTRUMENTATION

SOURCE RANGE MONITORS

SURVEILLANCE REQUIREMENTS

4.3.7.6 (Continued)

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

† For initial loading and startup the count rate may be less than 3 cps if the following conditions are met: (1) the signal-to-noise ratio is greater than or equal to 20 and (2) the signal is greater than 0.7 cps.

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITIONS FOR OPERATION

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

a. Five movable detectors, drives and readout equipment to map the core, and
b. Indexing equipment to allow all five detectors to be calibrated in a common location.

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

a. Recalibration of the LPRM detectors, and
b.* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE:

a. Within 72 hours before it is used for calibration of the LPRM detectors by normalizing each of the above required detector outputs.

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

c. At least once per 18 months by removing and detonating the explosive squib charge from one valve. The replacement charge for the exploded squib will be from the same manufactured batch as the one fired or from another batch that has been certified by having at least one charge from that batch successfully fired. All charges will be replaced in accordance with the manufacturer's recommendations. All explosive valves shall be tested once per 90 months.

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

INSTRUMENTATION

MONITORING INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.3.7.8 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

4.3.7.8 Each channel of the loose-part detection system shall be demonstrated OPERABLE by:

a. Performing a CHANNEL CHECK at least once per 24 hours,
b. Performing a CHANNEL FUNCTIONAL TEST at least once per 31 days,
c. Performing a CHANNEL CALIBRATION at least once per 18 months,
d. Listening to the audio output at least once per 7 days, and
e. Verifying acceptable background noise level at least once per 92 days.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.7.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.7.9-1 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: During releases via this pathway.

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, or change the setpoint so it is acceptably conservative.

b. With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3.7.9-1. Restore the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.7.9 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 at the frequencies shown in Table 4.3.7.9-1.

TABLE 3.3.7.9-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
Liquid Radwaste Effluent Line	1	128
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Service Water Effluent Line A	1	130
b. Service Water Effluent Line B	1	130
c. Cooling Tower Blowdown Line	1	130
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1	131
b. Service Water Effluent Line A	1	131
c. Service Water Effluent Line B	1	131
d. Cooling Tower Blowdown Line	1	131
4. Tank Level Indicating Devices*	1	132

* Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system, such as temporary tanks.

TABLE 3.3.7.9-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

ACTION 128 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that before initiating a release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 129 - Not used.

ACTION 130 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a limit of detection of at least 5×10^{-7} microcuries/ml.

ACTION 131 - 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 during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 132 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

TABLE 4.3.7.9-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
Liquid Radwaste Effluent Line	D	P	R(c)	M(a)(b)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Service Water Effluent Line A	D	M	R(c)	SA(b)
b. Service Water Effluent Line B	D	M	R(c)	SA(b)
c. Cooling Tower Blowdown Line	D	M	R(c)	SA(b)
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	D(d)	NA	R	Q
b. Service Water Effluent Line A	D(d)	NA	R	Q
c. Service Water Effluent Line B	D(d)	NA	R	Q
d. Cooling Tower Blowdown Line	D(d)	NA	R	Q
4. Tank Level Indicating Devices*	D**	NA	R	Q

* Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system, such as temporary tanks.

** During liquid additions to the tank.

TABLE 4.3.7.9-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

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

(1) Instrument indicates measured levels above the Alarm Setpoint, or
(2) Circuit failure, or
(3) Instrument indicates a downscale failure, or
(4) Instrument controls not set in operate mode.

(c) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards, standards that are traceable to the National Bureau of Standards, or using actual samples of liquid effluents that have been analyzed on a system that has been calibrated with National Bureau of Standards traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration may be used.

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

INSTRUMENTATION

MONITORING INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS FOR OPERATION

3.3.7.10 The radioactive gaseous 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 3.11.2.1 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 3.3.7.10-1.

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, or change the setpoint so it is acceptably conservative.

b. With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3.7.10-1. Restore the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.7.10 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 4.3.7.10-1.

TABLE 3.3.7.10-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Offgas System			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	136
b. System Flow-Rate Measuring Device	1	*	135
c. Sampler Flow-Rate Measuring Device	1	*	135
2. Offgas System Explosive Gas Monitoring System**			
a. Hydrogen Monitor Train A (Instrument 2OFG-AT-16A or 2OFG-AT-115)	1	*	137
b. Hydrogen Monitor Train B (Instrument 2OFG-AT-16B or 2OFG-AT-115)	1	*	137
3. Radwaste/Reactor Building Vent Effluent System			
a. Noble Gas Activity Monitor†	1	††	139
b. Iodine Sampler	1	††	138
c. Particulate Sampler	1	††	138
d. Flow-Rate Monitor	1	††	135
e. Sampler Flow-Rate Monitor	1	††	135
4. Main Stack Effluent			
a. Noble Gas Activity Monitor†	1	††	139
b. Iodine Sampler	1	††	138
c. Particulate Sampler	1	††	138
d. Flow-Rate Monitor	1	††	135
e. Sampler Flow-Rate Monitor	1	††	135

TABLE 3.3.7.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATIONS

* During offgas system operation.

** Only one train required to be in operation.

† Includes high range noble gas monitoring capability.

†† At all times.

ACTIONS

ACTION 135 - 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 136 - 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 137 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement operation of the offgas system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 138 - 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 starting within 8 hours of discovery, using auxiliary sampling equipment as required in Table 4.11.2-1.

ACTION 139 -

a. 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 for a radioactivity limit of detection of at least 1×10^{-4} microcurie/ml.

b. Restore the inoperable channel(s) to OPERABLE status within 72 hours or in lieu of another report required by Specification 6.9.1, 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 schedule for restoring the system to OPERABLE status.

TABLE 4.3.7.10-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Offgas System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	NA	R(a)	M(b,c)	**
b. System Flow-Rate Measuring Device	D	NA	R	Q	**
c. Sampler Flow-Rate Measuring Device	D	NA	R	Q	**
2. Offgas System Explosive Gas Monitoring System					
a. Hydrogen Monitor Train A	D	NA	Q(d)	M	**
b. Hydrogen Monitor Train B	D	NA	Q(d)	M	**
3. Radwaste/Reactor Building Vent Effluent System					
a. Noble Gas Activity Monitor†	D	M	R(a)	Q(c)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Flow-Rate Monitor	D	NA	R	Q	*
e. Sampler Flow-Rate Monitor	D	NA	R	Q	*

TABLE 4.3.7.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. Main Stack Effluent					
a. Noble Gas Activity Monitor †	D	M	R(a)	Q(c)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*
d. Flow-Rate Monitor	D	NA	R	Q	*
e. Sampler Flow-Rate Monitor	D	NA	R	Q	*

TABLE 4.3.7.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING

INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

* At all times.

** During offgas system operation.

† Includes high range noble gas monitoring capability.

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

(b) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

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

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

(d) 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 CONDITIONS FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

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

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:

   1. Four high-pressure turbine stop valves,
   2. Four high-pressure turbine control valves, and
   3. Six low-pressure turbine combined stop and intercept valves.

b. At least once per 18 months by direct observation of the movement of each of the above valves through at least one complete cycle from the running position, and by performance of a CHANNEL CALIBRATION of the turbine overspeed protection system.

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

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*The turbine overspeed protection system is not required to be OPERABLE prior to the initial opening of the main steam isolation valves in OPERATIONAL CONDITION 2.

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 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

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

b. With one or more Plant System Actuation Instrumentation channels inoperable take the ACTION required by Table 3.3.9-1.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3.9-1

PLANT SYSTEMS ACTUATION INSTRUMENTATION

TRIP FUNCTION	INSTRUMENT NUMBER	MINIMUM OPERABLE CHANNELS (a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Feedwater System/Main Turbine Trip System				
Reactor Vessel Water Level - High, Level 8	2ISC*LSH1624A,B,C	3	1	140
2. Service Water System				
a. Discharge Bay Level	2SWP*LS30A,B	2	1,2,3,4,5	142
b. Intake Tunnel 1 & 2 Water Temperature	2SWP*TSL64A,65A 2SWP*TSL64B,65B	1/Division 1/Division	1,2,3,4,5 1,2,3,4,5	144 144
c. Service Water Bay	2SWP*LS73A,B	2	1,2,3,4,5	143
d. Service Water Pumps Discharge Strainer Differential Pressure - Train "A"	2SWP*PDSH1A,C,E	1/Strainer	1,2,3,4,5	146
e. Service Water Pumps Discharge Strainer Differential Pressure - Train "B"	2SWP*PDSH1B,D,F	1/Strainer	1,2,3,4,5	146
f. Service Water Supply Header Discharge Water Temperature	2SWP*TY31A,B	2	1,2,3,4,5	147
g. Service Water Inlet Pressure for EDG*2 (HPCS, Division III)				
1) Division I Supply Header	2SWP*PSL95A	1	1,2,3,4,5	145
2) Division II Supply Header	2SWP*PSL95B	1	1,2,3,4,5	145

(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, except for discharge bay level and service water bay level which may be placed in an inoperable status for up to 4 hours without placing the Trip System in a tripped condition.

TABLE 3.3.9-1 (Continued)

PLANT SYSTEMS ACTUATION INSTRUMENTATION

ACTION

ACTION 140 - a. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

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

ACTION 141 - Not used.

ACTION 142 - Monitor discharge bay level continuously if level reaches trip setpoint, provide an alternate flow discharge path by locking closed 2SWP*MOV30A or 2SWP*MOV30B.

ACTION 143 - Monitor service water bay level continuously if level reaches Trip Setpoint provide an alternate intake to the service bay by locking open 2SWP*MOV77A or 2SWP*MOV77B.

ACTION 144 - Place intake heaters in service if lake temperature $\leq$ 39°F or take the ACTIONS required by Specifications 3.7.1.1 and 3.7.1.2, as appropriate.

ACTION 145 - Lock closed 2SWP*MOV95A or 2SWP*MOV95B and declare EDG-2 (HPCS, Division III) inoperable and take the ACTION required by Specification 3.8.1.

ACTION 146 - Monitor the effected pump discharge pressure and the applicable service water loop header pressure to determine the differential pressure across the strainer; if the differential pressure exceeds the setpoint manually start the strainer or declare the effected service water pump inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2, as appropriate.

ACTION 147 - Monitor service water local discharge temperature indicators as applicable per Specification 4.7.1.1.1.a.2 or 4.7.1.2.1.a.2.

TABLE 3.3.9-2

PLANT SYSTEMS ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Feedwater System/Main Turbine Trip System		
a. Reactor Vessel Water Level - High Level 8	≤202.3 in.*	≤209.3 in.
2. Service Water System		
a. Discharge Bay Level	≤275' Elev.	≤275' 2-3/4" Elev.
b. Intake Tunnel 1 & 2 Water Temperature	≥39°F	≥38°F
c. Service Water Bay	≥234' Elev.	≥233' 1-1/4" Elev.
d. Service Water Pumps Discharge Strainer Differential Pressure - Train "A"	≤10 psid	≤14.5 psid
e. Service Water Pumps Discharge Strainer Differential Pressure-Train "B"	≤10 psid	≤14.5 psid
f. Service Water Supply Header Discharge Water Temperature	NA	NA
g. Service Water Inlet Pressure for EDG*2 (HPCS, Division III)		
1) Division I Supply Header	≥25 psig	≥17.5 psig
2) Division II Supply Header	≥22 psig	≥17.5 psig

*See Bases Figure B3/4 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 FOR WHICH SURVEILLANCE IS REQUIRED
1. Feedwater System/Main Turbine Trip System				
a. Reactor Vessel Water Level - High Level 8	NA	M	R	1
2. Service Water System				
a. Discharge Bay Level	NA	R	R	1, 2, 3, 4, 5
b. Intake Tunnel 1 & 2 Water Temperature	W	R	R*	1, 2, 3, 4, 5
c. Service Water Bay	NA	R	R	1, 2, 3, 4, 5
d. Service Water Pumps Discharge Strainer Differential Pressure - Train "A"	S	R	R	1, 2, 3, 4, 5
e. Service Water Pumps Discharge Strainer Differential Pressure - Train "B"	S	R	R	1, 2, 3, 4, 5
f. Service Water Supply Header Discharge Water Temperature	S	R	R	1, 2, 3, 4, 5
g. Service Water Inlet Pressure for EDG*2 (HPCS, Division III)				
1) Division I Supply Header	NA	R	R	1, 2, 3, 4, 5
2) Division II Supply Header	NA	R	R	1, 2, 3, 4, 5

* Calibration excludes sensors; a comparison test of the four RTDs will be done.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITIONS FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or
b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within four hours:

      a) Place the recirculation flow control system in the Loop Manual (Position Control) mode, and

      b) Reduce THERMAL POWER to $\leq$ 70% of RATED THERMAL POWER, and,

      c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.07 per Specification 2.1.2, and,

      d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value of 0.81 times the two recirculation loop operation limit per Specification 3.2.1, and,

      e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1, 3.2.2 and 3.3.6.

      f) Reduce the volumetric flow rate of the operating recirculation loop to $\leq$ 41,000** gpm.

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* See Special Test Exception 3.10.4.

** This value represents the design volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. The actual value will be established during the Startup Test Program.

LIMITING CONDITIONS FOR OPERATION (Continued)

g) Perform Surveillance Requirement 4.4.1.1.2 if THERMAL POWER is $\leq$ 30%* of RATED THERMAL POWER or the recirculation loop flow in the operating loop is $\leq$ 50%* of rated loop flow.

2. The provisions of Specification 3.0.4 are not applicable.

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

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER such that it is not within the restricted zone of Figure 3.4.1.1-1 within two hours, and initiate measures to place the unit in at least STARTUP within six hours and in HOT SHUTDOWN within the next six hours.

c. With one or two reactor coolant system recirculation loops in operation and total core flow less than 45% but greater than 39%** of rated core flow and THERMAL POWER within the restricted zone of Figure 3.4.1.1-1:

1. Determine the APRM and LPRM*** noise levels per Specification 4.4.1.1.4:

   a) At least once per eight hours, and

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

2. With the APRM or LPRM*** neutron flux noise levels greater than three times their established baseline noise levels, within 15 minutes initiate corrective action to restore the noise levels within the required limits within two hours by increasing core flow or by reducing THERMAL POWER.

d. With one or two reactor coolant system recirculation loops in operation and total core flow $\leq$ 39%** and THERMAL POWER within the restricted zone of Figure 3.4.1.1-1, within 15 minutes initiate corrective action to reduce THERMAL POWER to within the unrestricted zone of Figure 3.4.1.1-1 or increase core flow to > 39%** within 4 hours.

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* Initial values. Final values to be determined during Startup Testing based upon the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

** Value to be established during startup test program which is equivalent to minimum core flow for 2 recirculation pumps at high speed with minimum flow control valve position.

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

REACTOR COOLANT SYSTEM

RECIRCULATION SYSTEM

RECIRCULATION LOOPS

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

a. Reactor THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Loop Manual (Position Control) mode,

c. The volumetric flow rate of the operating loop is $\leq$ 41,000 gpm,* and

d. Core flow is > 39%** when THERMAL POWER is within the restricted zone of Figure 3.4.1.1-1.

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $\leq$ 30%*** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is $\leq$ 50%*** of rated loop flow:

a. $\leq$ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. $\leq$ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. $\leq$ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.2 b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

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* This value represents the design volumetric recirculation loop flow which produces 100% core flow at 100% THERMAL POWER. The actual value will be established during the Startup Test Program.

** Value to be established during startup test program which is equivalent to minimum core flow for 2 recirculation pumps at high speed with minimum flow control valve position.

*** Initial values. Final values to be determined during Startup Testing based upon the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

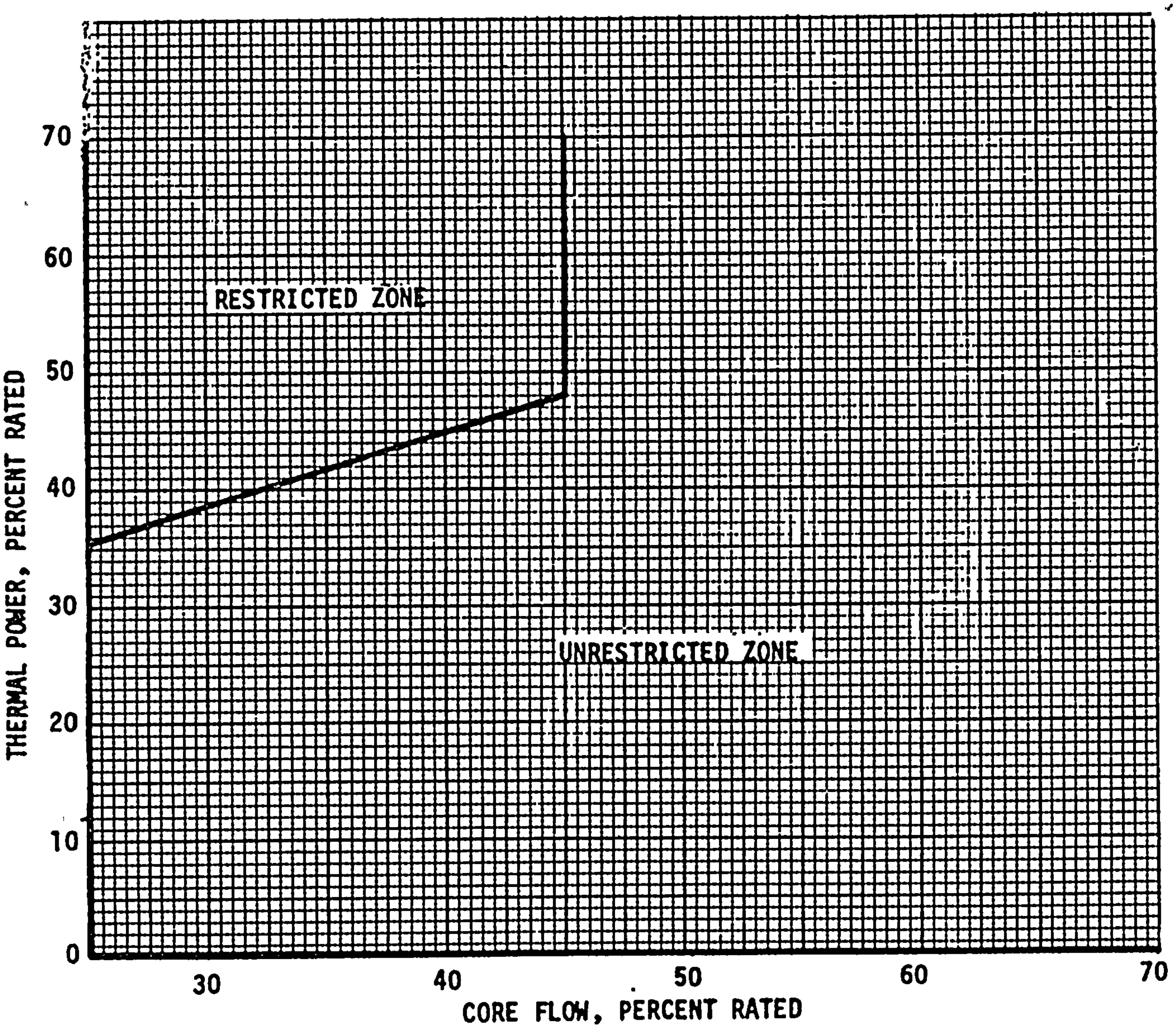

Figure 3.4.1.1-1 Percent of Rated Core Thermal Power vs. Percent of Rated Core Flow

REACTOR COOLANT SYSTEM

RECIRCULATION SYSTEM

JET PUMPS

LIMITING CONDITIONS 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 All jet pumps shall be demonstrated operable as follows:

a. Each of the above required jet pumps shall be demonstrated OPERABLE before THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours while greater than 25% of RATED THERMAL POWER, by determining recirculation loop flow, total core flow, and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.

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

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

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

b. During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur:

   1. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* single recirculation loop control valve position-loop flow characteristics.

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

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* To be determined during the startup test program.

REACTOR COOLANT SYSTEM

RECIRCULATION SYSTEM

JET PUMPS

SURVEILLANCE REQUIREMENTS (Continued)

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

c. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

* To be determined during the startup test program.

REACTOR COOLANT SYSTEM

RECIRCULATION SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITIONS FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 during two recirculation loop operation.**

ACTION:

With the 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. Shut down one of the recirculation loops 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.

* Effective core flow shall be the core flow that would result if both recirculation loop flows were assumed to be at the smaller value of the two loop flows.

** See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

RECIRCULATION SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITIONS FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F, and:

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITIONS FOR OPERATION

3.4.2 The safety valve function of at least 16 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings*; the acoustic monitor for each OPERABLE valve shall be OPERABLE:

2 safety/relief valves @ 1148 psig ±1%
4 safety/relief valves @ 1175 psig ±1%
4 safety/relief valves @ 1185 psig ±1%
4 safety/relief valves @ 1195 psig ±1%
4 safety/relief valves @ 1205 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required 16 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 the average water temperature in the suppression pool is less than 110°F, close the stuck-open safety/relief valve(s); if unable to close the open valve(s) within 5 minutes or if the average water temperature in the suppression pool is 110°F or more, 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.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.25 of the full-open noise level* by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months.**

*Initial setting shall be in accordance with the manufacturers recommendation. Adjustment to the valve full-open noise level shall be accomplished during the startup test program.

**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.

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

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

a. The primary containment airborne particulate radioactivity monitoring system,

b. The primary containment drywell floor drain tank and equipment drain tank fill rate monitoring systems, and

c. The primary containment airborne 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 containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

a. Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a SOURCE CHECK at least once per 31 days, a CHANNEL FUNCTIONAL TEST at least once per 184 days and a CHANNEL CALIBRATION at least once per 18 months.

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

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.4.3.2 Reactor coolant system (RCS) leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.
b. 5 gpm UNIDENTIFIED LEAKAGE.
c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period.
d. 0.5 gpm leakage per nominal inch of valve size up to a maximum 5 gpm at an RCS pressure of 1020 ± 20 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

b. With any RCS leakage greater than the limits in Specification 3.4.3.2.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 RCS 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 or 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 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With one or more of the required interlocks shown in Table 3.4.3.2-3 inoperable, restore the inoperable interlock to OPERABLE status within 7 days or isolate the affected heat exchanger(s) from the RCIC steam supply by closing and deenergizing heat exchanger valves 2 RHS*MOV22A and 2RHS*MOV80A or 2RHS*MOV22B and 2RHS*MOV80B, as appropriate.

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* Which have been verified not to exceed the allowable leakage limit at the last refueling outage.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The RCS leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment airborne particulate radioactivity at least once per 12 hours,

b. Monitoring the primary containment drywell floor drain tank and equipment drain tank fill rate at least once per 12 hours,

c. Monitoring the primary containment airborne 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 RCS pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 as outlined in the ASME Code Section XI, paragraph IWV-3427(b) and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and
b. Before returning the valve to service following maintenance, repair, or replacement work on the valve.

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 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.

4.4.3.2.4 The high/low-pressure interface interlock for the steam condensing mode bypass valve shall be demonstrated OPERABLE with trips setpoints per Table 3.4.3.2-3 by performance of:

a. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
b. CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
2CSH*MOV107	HPCS
2CSH*AOV108	HPCS
2CSL*MOV104	LPCS
2CSL*AOV101	LPCS
2ICS*AOV156	RCIC
2ICS*AOV157	RCIC
2RHS*MOV112	RHR
2RHS*MOV113	RHR
2RHS*MOV104	RHR
2RHS*MOV40 A&B	RHR
2RHS*MOV67 A&B	RHR
2RHS*MOV24 A,B&C	RHR-(LPCI)
2RHS*AOV16 A,B&C	RHR-(LPCI)
2RHS*AOV39 A&B	RHR
2RHS*MOV22 A&B	RHR
2RHS*MOV23 A&B	RHR
2RHS*MOV80 A&B	RHR

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES - LEAKAGE PRESSURE MONITORS

INSTRUMENT NUMBER	VALVE NUMBER	SETPOINT (PSIG)
2RHS*PSX7A†	2RHS*MOV24A	475 ±6
	2RHS*MOV40A	475 ±6
	2RHS*MOV22A	475 ±6
	2RHS*MOV23A	475 ±6
	2RHS*MOV80A	475 ±6
2RHS*PSX7B†	2RHS*MOV24B	475 ±6
	2RHS*MOV40B	475 ±6
	2RHS*MOV22B	475 ±6
	2RHS*MOV23B	475 ±6
	2RHS*MOV80B	475 ±6
	2RHS*MOV104	475 ±6
2RHS*PSX7C†	2RHS*MOV24C	475 ±6
2CSL*PS108	2CSL*MOV104	525 ±6
2RHS*PIS111	2RHS*MOV112	171 ±6
	2RHS*MOV113	171 ±6

TABLE 3.4.3.2-3

HIGH/LOW-PRESSURE INTERFACE INTERLOCKS

INSTRUMENT NUMBER	VALVE NUMBER	SETPOINT (PSIG)
2RHS*PS75A/76A†	2RHS*MOV23A	465 ±12
2RHS*PS75B/76B†	2RHS*MOV23B	465 ±12

†Pressure switch has process indication.

REACTOR COOLANT SYSTEM

3/4.4.4 CHEMISTRY

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With the conductivity, chloride concentration, or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration for less than 336 hours per year, but with the conductivity less than 10 µmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.4 are not applicable.

2. With the conductivity, chloride concentration, or pH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

3. With the conductivity exceeding 10 µmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITIONS 2 and 3 with the conductivity, chloride concentration, or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

1. With the:

a) Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours, or

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

LIMITING CONDITIONS FOR OPERATION

3.4.4 (Continued)

ACTION:

c.1.b) (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 before 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 before 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.

SURVEILLANCE REQUIREMENTS

4.4.4 (Continued)

d. Performing 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 (μmho/cm @25°C)	pH
1	$\leq$0.2 ppm	$\leq$1.0	$5.6 \leq pH \leq 8.6$
2 and 3	$\leq$0.1 ppm	$\leq$2.0	$5.6 \leq pH \leq 8.6$
At all other times	$\leq$0.5 ppm	$\leq$10.0	$5.3 \leq pH \leq 8.6$

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITIONS 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 DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131,

   2. 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 4.a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

c. In OPERATIONAL CONDITION 1 or 2, with:

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

   2. The offgas level, downstream of the recombiner, increased by more than 10,000 microcuries per second in 1 hour during steady-state operation at release rates less than 75,000 microcuries per second, or

   3. The offgas level, downstream of the recombiner, increased by more than 15% in 1 hour during steady-state operation at release rates greater than 75,000 microcuries per second,

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

* Not applicable during the startup test program

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

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 ARE 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 offgas 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 CONDITIONS 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 for hydrostatic or leak testing; Figure 3.4.6.1-2 for heatup by non-nuclear means, cooldown following a nuclear shutdown and low-power PHYSICS TESTS; and Figure 3.4.6.1-3 for operations with a critical core other than low-power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

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

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 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 Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3, as applicable, at least once per 30 minutes.

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

SURVEILLANCE REQUIREMENTS

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 3.4.6.1-3 within 15 minutes before 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 bring up to date the curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3.

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 when reactor vessel head bolting studs are under full tension:

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

    1. ≤90°F, at least once per 12 hours.
    2. ≤80°F, at least once per 30 minutes.

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

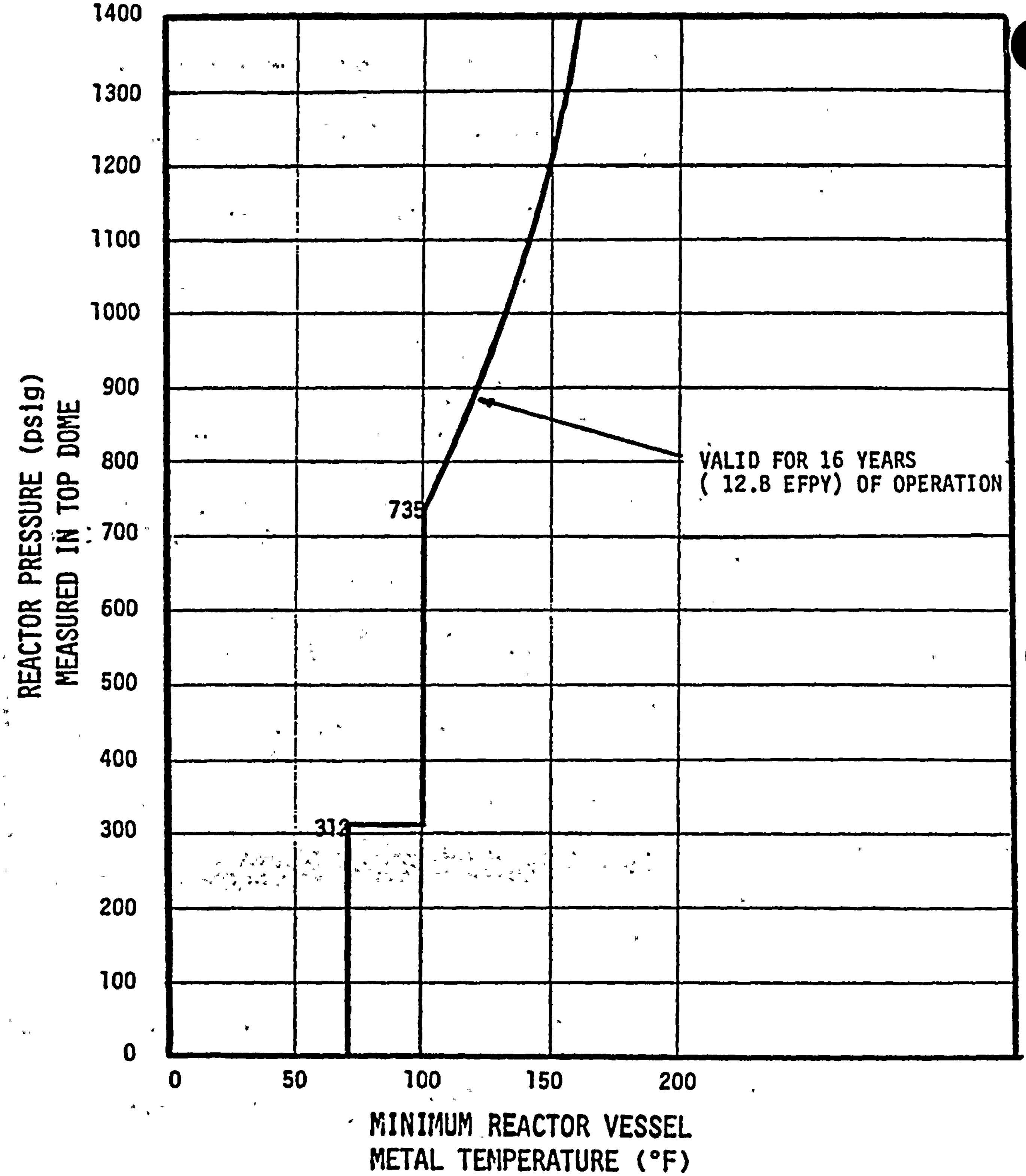

Figure 3.4.6.1-1 Minimum Reactor Vessel Temperature for Pressurization During In-Service Hydrostatic Testing and Leak Testing

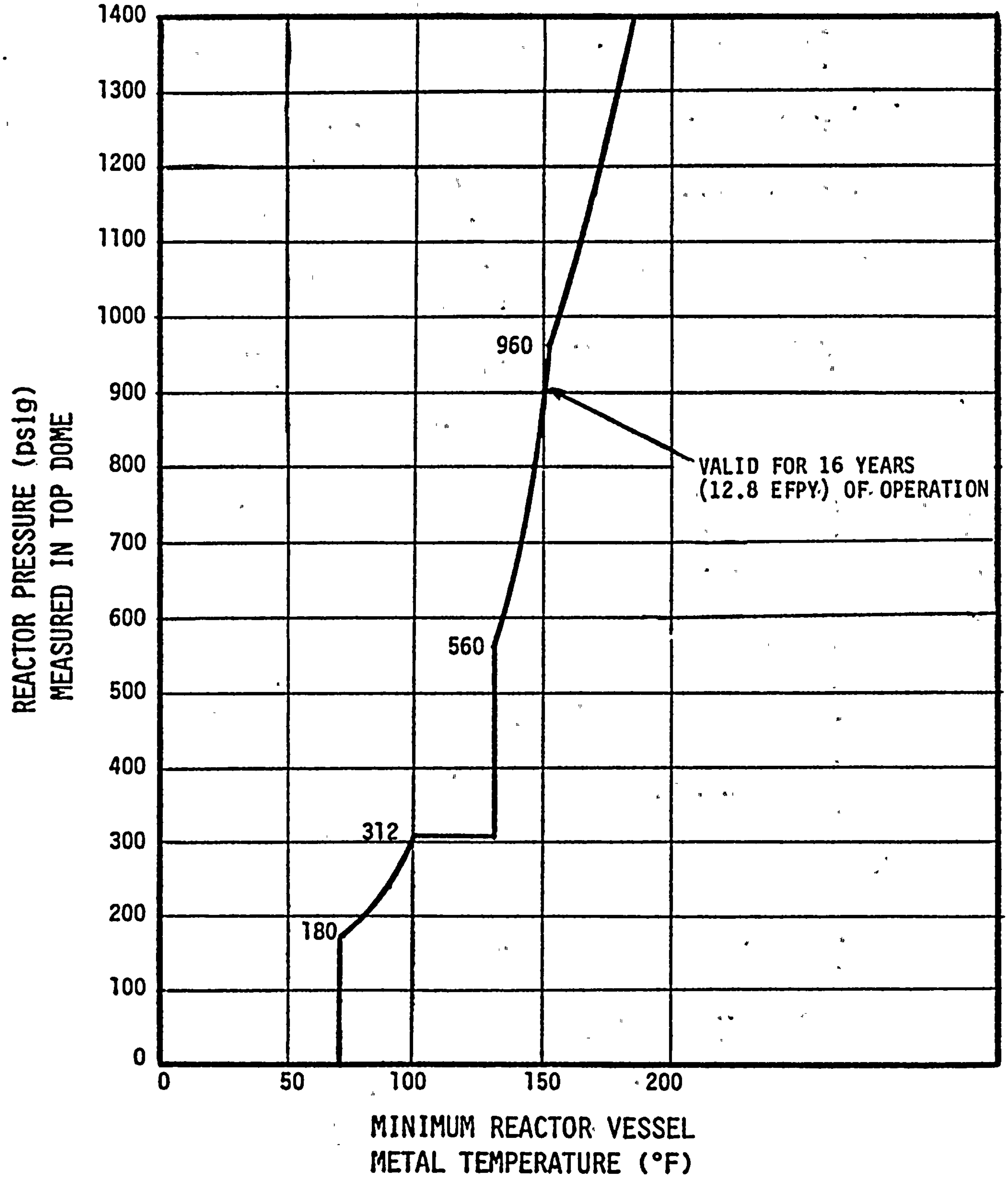

Figure 3.4.6.1-2 Minimum Reactor Vessel Temperature for Pressurization During Non-Nuclear Heatup/ Cooldown and Low-Power Physics Tests

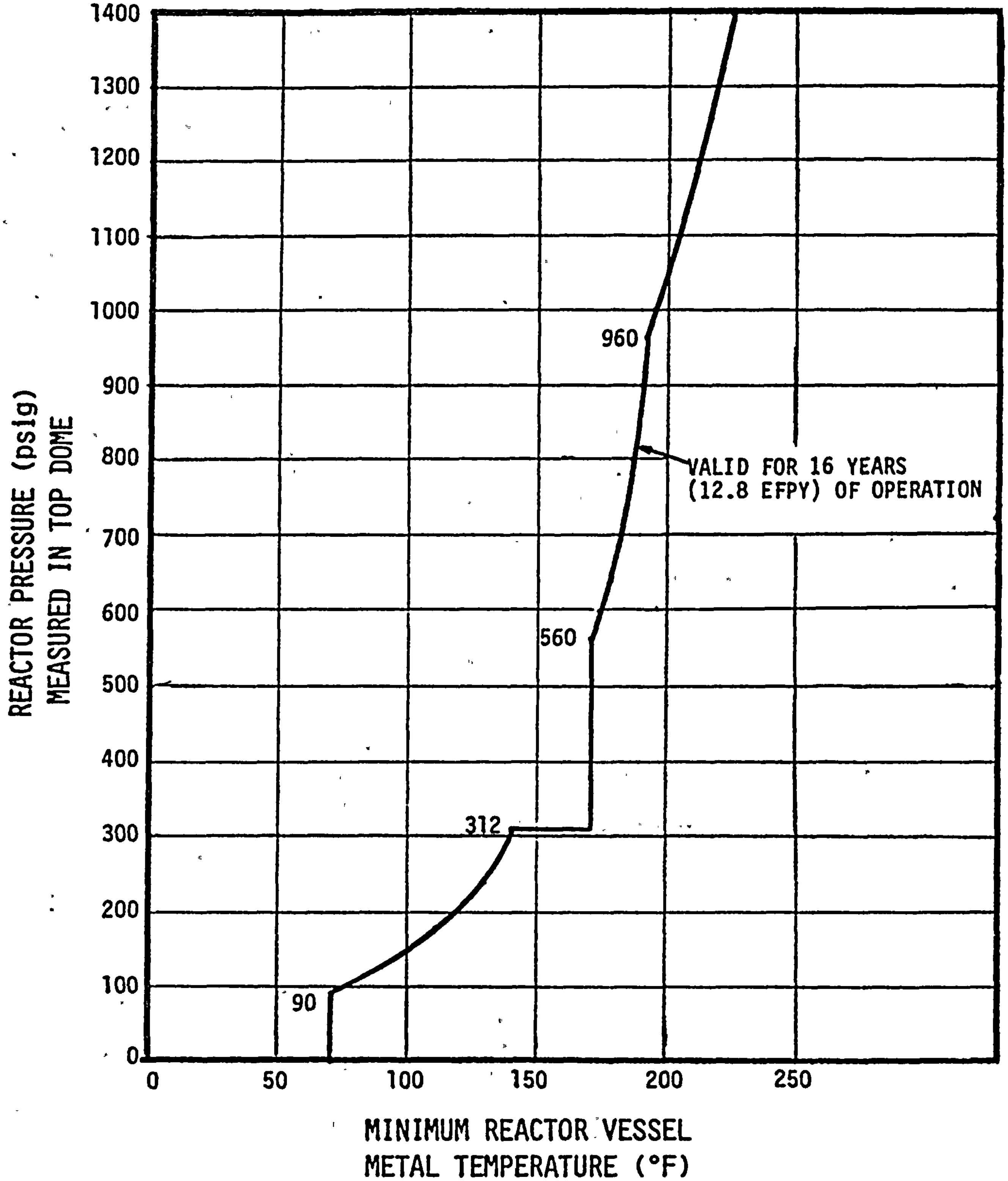

Figure 3.4.6.1-3 Minimum Reactor Vessel Temperature for Pressurization During Core Critical Operation

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ 1/4 T	WITHDRAWAL TIME (EFPY)
1	3°	0.41	10
2	177°	0.41	20
3	183°	0.41	Spare

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

REACTOR STEAM DOME

LIMITING CONDITIONS FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1020 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 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 1020 psig at least once per 12 hours.

* Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITIONS 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 4 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.

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITIONS 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) before 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) before 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.

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

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

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

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

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop. Be in at least COLD SHUTDOWN within 24 hours.††

b. With no RHR shutdown cooling mode loop in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within 1 hour, establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

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

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* 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.

REACTOR COOLANT SYSTEM

RESIDUAL HEAT REMOVAL

COLD SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

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

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION:

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system or alternative 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 every 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 CONDITIONS FOR OPERATION

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

a. ECCS Division I consisting of:

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

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

3. Seven OPERABLE ADS valves.

b. ECCS Division II consisting of:

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

2. Seven OPERABLE ADS valves.

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

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

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* 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 3.10.6.

† LPCI subsystems of the RHR system may be inoperable in that they are aligned in the shutdown cooling mode when reactor vessel pressure is less than the RHR shutdown cooling permissive setpoint.

LIMITING CONDITIONS FOR OPERATION

3.5.1 (Continued)

ACTION:

a. For ECCS Division I, provided that ECCS Divisions II and III are OPERABLE:

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

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

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

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

b. For ECCS Division II, provided that ECCS Divisions I and III are OPERABLE:

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

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

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

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

   1. With ECCS Division III inoperable, 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.

LIMITING CONDITIONS FOR OPERATION

3.5.1 (Continued)

ACTION:

d. For ECCS Divisions I and II, provided that ECCS Division III is OPERABLE:

   1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or the 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 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*.

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

   1. With up to two of the above required ADS valves inoperable, restore the inoperable ADS valve(s) to OPERABLE status within 14 days of the first ADS valve becoming inoperable or 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 next 24 hours.

   2. With three 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 less than or equal to 100 psig within the next 24 hours.

f. In the event an ECCS 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 usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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* 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 I, II and III shall be demonstrated OPERABLE by:

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

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

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

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

   1. LPCS pump develops a flow of at least 6350 gpm against a test line pressure greater than or equal to 290 psig.

   2. LPCI pump develops a flow of at least 7450 gpm against a test line pressure greater than or equal to 134 psig for loops A and B, and greater than or equal to 147 psig for loop C.

   3. HPCS pump develops a flow of at least 6350 gpm against a test line pressure greater than or equal to 333 psig.

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.

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* 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.

† Verify HPCS pump will auto-restart on low reactor vessel water level, level 2, if the pump has been manually stopped.

SURVEILLANCE REQUIREMENTS

4.5.1 (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, 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 that either:

      1) The SRV discharge acoustic monitoring system responds accordingly, or

      2) The control valve or bypass valve responds accordingly, or

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

      4) The SRV discharge line temperature monitoring system responds accordingly.

   c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system, low-pressure alarm system, and verifying an alarm setpoint of 163.5 + 3.2, -3.2 psig on decreasing pressure.

   d) Performing a leak rate test for ADS SRV pneumatic operators by pressurizing each ADS accumulator at 178 psig (supply header high pressure alarm) up to its supply header isolation check valve with the SRV in the open position. Total leakage rate for each SRV shall not exceed 0.5 SCFH for the SRV actuated by either of the ADS solenoids.

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

SURVEILLANCE REQUIREMENTS

4.5.1 (Continued)

e) Performing a leak rate test for the safety related ADS accumulator pneumatic supply system (including special emergency tube trailer supply piping) up to SRV actuators/operators. With the SRV's actuated by either of the ADS solenoids and with ADS accumulators at 178 psig and with ADS nitrogen receiving tanks at 385 psig (high pressure alarm), the leakage rates shall not exceed the following limits:

1. For the ADS SRV actuators, supply header and accumulators, and the nitrogen receiving tank for the SRV's 2MSS* PSV 121, 126, & 127, maximum allowable leakage is 3 SCFH.

2. For the ADS SRV actuators, supply header and accumulators, and the nitrogen receiving tank for the SRV's 2MSS* PSV 129, 130, 134, & 137, maximum allowable leakage is 4 SCFH.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITIONS 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 chamber 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 chamber 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 chamber 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 chamber 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 chamber, or
   2. When the suppression pool level is less than the limit or is drained, from the "B" condensate storage tank containing at least 253,000 available gallons of water, equivalent to a level of 26.9 feet.

APPLICABILITY: OPERATIONAL CONDITIONS 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 spent fuel pool gates are removed, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

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

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

3/4.5.3 SUPPRESSION POOL

LIMITING CONDITIONS FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

a. In OPERATIONAL CONDITIONS 1, 2, and 3 with a contained water volume of at least 145,495 cubic feet, equivalent to an elevation of 199 feet 6 inches.

b. In OPERATIONAL CONDITIONS 4 and 5* with a contained water volume of at least 145,495 cubic feet, equivalent to a level of 199 feet 6 inches 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 "B" condensate storage tank contains at least 253,000 available gallons of water, equivalent to a level of 26.9 feet, and

   4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the "B" 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 in OPERATIONAL CONDITION 5, provided that the reactor vessel head is removed, the cavity is flooded or being flooded, the spent fuel pool gates are removed (when the cavity is flooded), and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool shall be determined OPERABLE by verifying the water level to be greater than or equal to 199 feet 6 inches at least once every 24 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

b. Verify footnote conditions * to be satisfied.

* The suppression pool is not required to be OPERABLE in OPERATIONAL CONDITION 5 provided that the reactor vessel head is removed, the cavity is flooded or being flooded, 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 CONDITIONS FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at Pa, 39.75 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

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

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

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

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* See Special Test Exception 3.10.1

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

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to:

   1. La, 1.1% by weight of the containment air every 24 hours at Pa, 39.75 psig, or

   2. Lt, 0.72% by weight of the containment air every 24 hours at a reduced pressure of Pt, 20.0 psig.

b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests when pressurized to Pa, 39.75 psig.

c. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 Pa, 43.73 psig.

d. Less than or equal to that specified in Table 3.6.1.2-1 through valves in lines that are potential bypass leakage pathways when tested at 40.0 psig.

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

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 La or 0.75 Lt, as applicable, or

* Exemption to Appendix J of 10 CFR 50.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.6.1.2 (Continued)

ACTION:

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

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

d. The measured leakage rate through any valve that is part of a potential bypass leakage pathway exceeding the limit specified in Table 3.6.1.2-1

Restore:

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

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

c. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which pentrate the primary containment to less than or equal to 1 gpm times the total number of such valves, and

d. The leakage rate to less than or equal to that specified in Table 3.6.1.2-1 for any valve that is part of a potential bypass leakage path.

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

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* Exemption to Appendix J to 10 CFR 50.

SURVEILLANCE REQUIREMENTS

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

a. Three Type A overall integrated containment leakage rate tests shall be conducted at 40 ± 10-month intervals during shutdown at Pa, 39.75 psig or at Pt, 20.0 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 La or 0.75 Lt, as applicable, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La or 0.75 Lt, as applicable, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La or 0.75 Lt, as applicable, 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 La or 0.25 Lt, as applicable.

    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% of the total measured leakage at Pa, 39.75 psig, or Pt, 20.0 psig, as applicable.

d. Type B and C tests shall be conducted with gas at Pa, 39.75 psig,* at intervals no greater than 24 months except for tests involving:

    1. Air locks,

    2. Main steam line isolation valves and the remainder of the valves specified in Table 3.6.1.2-1.

    3. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

    4. Purge supply and exhaust isolation valves with resilient seals.

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

PRIMARY CONTAINMENT LEAKAGE

SURVEILLANCE REQUIREMENTS

4.6.1.2 (Continued)

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

f. Main steam line isolation valves and the remainder of the valves specified in Table 3.6.1.2-1 shall be leak tested at least once per 18 months.

g. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at Pa, 39.75 psig, at intervals no greater than once per 3 years.

h. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J. Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 Pa, 43.73 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

i. Containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

j. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.7.2.

k. The provisions of Specification 4.0.2 are not applicable to Surveillance Requirements 4.6.1.2.a and 4.6.1.2.d.

TABLE 3.6.1.2-1

ALLOWABLE LEAK RATES THROUGH VALVES IN

POTENTIAL BYPASS LEAKAGE PATHS

LINE DESCRIPTION	VALVE MARK NO	TERMINATION REGION	PER VALVE* LEAK RATE, SCFH
4 Main Steam Lines	2MSS*AOV6A, B, C, D 2MSS*AOV7A, B, C, D	Turbine Bldg.	6.0
Main Steam Drain Line (Inboard)	2MSS*MOV111, 112	Turbine Bldg.	1.875
Main Steam Drain Line (Outboard)	2MSS*MOV208	Turbine Bldg.	0.625
4 Postaccident Sampling Lines	2CMS*SOV77A, B 2CMS*SOV74A, B 2CMS*SOV75A, B 2CMS*SOV76A, B	Radwaste Tunnel	0.2344
Drywell Equipment Drain Line	2DER*MOV119 2DER*MOV120	Radwaste Tunnel	1.25
Drywell Equipment Vent Line	2DER*MOV130 2DER*MOV131	Radwaste Tunnel	0.625
Drywell Floor Drain Line	2DFR*MOV120 2DFR*MOV121	Radwaste Tunnel	1.875
Drywell Floor Vent Line	2DFR*MOV139 2DFR*MOV140	Radwaste Tunnel	0.9375
RWCU Line	2WCS*MOV102 2WCS*MOV112	Turbine Bldg.	2.5
Feedwater Line	2FWS*AOV23A 2FWS*V12A 2FWS*AOV23B 2FWS*V12B	Turbine Bldg.	12.0
CPS Supply Line to Drywell	2CPS*AOV104 2CPS*AOV106	Standby Gas Trtmt. Area	4.38
CPS Supply Line to Drywell	2CPS*SOV120 2CPS*SOV122	Standby Gas Trtmt. Area	0.625
CPS Supply Line to Supp. Chamber	2CPS*AOV105 2CPS*AOV107	Standby Gas Trtmt. Area	3.75
CPS Supply Line to Supp. Chamber	2CPS*SOV119 2CPS*SOV121	Standby Gas Trtmt. Area	0.625

* Test conditions: air medium, 40 psig.

TABLE 3.6.1.2-1 (Continued)

ALLOWABLE LEAK RATES THROUGH VALVES IN

POTENTIAL BYPASS LEAKAGE PATHS

LINE DESCRIPTION	VALVE MARK NO	TERMINATION REGION	PER VALVE* LEAK RATE, SCFH
Inst. Air to ADS Valve Accumulator	IAS*SOV164 IAS*V448	Yard Area	0.9375
Inst. Air to ADS Valve Accumulator	IAS*SOV165 IAS*V449	Yard Area	0.9375
N_2 Purge to TIP Index Mechanism	GSN*SOV166 GSN*V170	Yard Area	**
Inst. Air to SRV Accumulator	IAS*SOV166 IAS*SOV184	Yard Area	**
Inst. Air to Drywell	IAS*SOV167 IAS*SOV185	Yard Area	**
Inst. Air to Drywell	IAS*SOV168 IAS*SOV180	Yard Area	**
Inst. Air to CPS Valve in Suppression Chamber	CPS*SOV132 CPS*V50	Yard Area	**
Inst. Air to CPS Valve in Suppression Chamber	CPS*SOV133 CPS*V51	Yard Area	**

*Test Conditions - Air Medium, 40 psig.
**The combined leakage of these six penetrations shall not exceed 3.6 SCFH. The leakage through each penetration shall be that of the valve with the highest rate in that penetration.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITIONS FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 La at Pa, 39.75 psig.

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

ACTION:

a. With one primary containment air lock door inoperable:

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

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

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

   4. The provisions of Specification 3.0.4 are not applicable.

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

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to greater than or equal to 10 psig:

   1. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours*; and

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used and no maintenance has been performed on the air lock that could affect the air lock sealing capability.

b. By conducting an overall air lock leakage test at Pa, 39.75 psig and by verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months*, and

   2. Before establishing PRIMARY CONTAINMENT INTEGRITY when maintenance had 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.†

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* The provisions of Specification 4.0.2 are not applicable.

** Exemption to Appendix J of 10 CFR 50.

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

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment, including the liner plate and drywell to wetwell bypass paths, 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 before the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.4.2 Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the vessel and the annulus fill concrete, the inspection procedure, the tolerances on concrete cracking, and the corrective actions taken.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITIONS FOR OPERATION

3.6.1.5 Drywell and suppression chamber internal pressure shall be maintained between 14.2 and 15.45 psia.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell and suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 Drywell average air temperature shall not exceed 150°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 150°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

4.6.1.6 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
306'-9"	354°
294'-5"	117°
283'-0"	58°
268'-0"	203°
255'-6"	326°
244'-0"	284°
306'-9"	189°
296'-4"	323°
282'-6"	243°
262'-3"	28°
253'-11"	169°
244'-0"	110°

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PURGE SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.1.7 The drywell and suppression chamber 12-inch and 14-inch purge supply and exhaust isolation valves shall be OPERABLE and:

a. The 12-inch (2CPS*AOV105, 2CPS*AOV107, 2CPS*AOV109, 2CPS*AOV111) and 14-inch (2CPS*AOV104, 2CPS*AOV106, 2CPS*AOV108, 2CPS*AOV110) valves in the purge system supply and exhaust lines may be open for up to 90 hours per 365 days for VENTING or PURGING.*

b. Purge system valves 2CPS*AOV105 (12-inch), 2CPS*AOV107 (12-inch), 2CPS*AOV109 (12-inch), and 2CPS*AOV110 (14-inch) shall be blocked to limit the opening to 70°. Purge system valve 2CPS*AOV111 (12-inch) shall be blocked to limit the opening to 60°.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the drywell and suppression chamber purge supply and/or exhaust isolation valve(s) inoperable, or open for more than 90 hours per 365 days for other than pressure control*, close the open valve(s); otherwise isolate the penetration(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a drywell and suppression chamber purge supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.7.2, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 At least once per refueling outage each drywell and suppression chamber purge supply and exhaust isolation valve of Specification 3.6.1.7.b shall be verified to be blocked to limit the opening to 70° or 60°, as applicable.

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* The 90-hour limit shall not apply to the use of valves 2CPS*AOV108 (14-inch) and 2CPS*AOV110 (14-inch), or 2CPS*AOV109 (12-inch) and 2CPS*AOV111 (12-inch), for primary containment pressure control, provided 2GTS*AOV101 is closed, and its 2-inch bypass line is the only flow path to the standby gas treatment system.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PURGE SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.1.7.2 At least once per 92 days each 12- and 14-inch drywell and suppression chamber purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 4.38 scf per hour per 14-inch valve and 3.75 scf per hour per 12-inch valve when pressurized to Pa, 39.75 psig. Those purge supply and exhaust isolation valves listed on Table 3.6.1.2-1 shall be pressurized to 40.0 psig.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL

LIMITING CONDITIONS FOR OPERATION

3.6.2.1 The suppression pool shall be OPERABLE with:

a. The pool water:

   1. Volume between 154,794 cubic feet and 145,495 cubic feet equivalent to an elevation between 201 feet and 199 feet 6 inches and a

   2. Maximum average temperature of 90°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

      a) 105°F during testing that adds heat to the suppression pool.

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

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

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{K}$ design value of 0.054 square feet.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

b. In OPERATIONAL CONDITION 1 or 2 with the suppression pool average water temperature greater than 90°F, restore the average temperature to less than or equal to 90°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 90°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

3.6.2.1.b (Continued)

ACTION:

2. With the suppression pool average water temperature greater than:

   a) 90°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.

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

CONTAINMENT SYSTEMS

DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL

SURVEILLANCE REQUIREMENTS

4.6.2.1 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 90°F, except:

   1. During testing that adds heat to the suppression pool verify the suppression pool average water temperature to be less than or equal to 105°F at least once per 5 minutes.

   2. When suppression pool average water temperature is greater than or equal to 90°F, verify at least once per hour that:

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

      b) THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 90°F for more than 24 hours.

   3. Following a scram with suppression pool average water temperature greater than 90°F, verify suppression pool average water temperature to be less than or equal to 120°F at least once per 30 minutes.

c. By verifying at least 20 suppression pool water temperature instrumentation channels* OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,
   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
   3. CHANNEL CALIBRATION** at least once per 18 months,

   with the water high temperature alarm setpoints $\leq$ 90°F for 10 of the temperature instruments and $\leq$ 110°F for 10 of the temperature instruments.

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* At least one pair in each of 10 suppression pool sectors with the alarm setpoint alternating between adjacent sectors.

** Calibration excludes sensors; sensors comparisons shall be made in lieu of calibration.

SURVEILLANCE REQUIREMENTS

4.6.2.1 (Continued)

d. At least once per 18 months by

   1. Conducting a visual inspection of the exposed accessible interior and exterior surfaces of the suppression chamber.*

   2. Conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 3 psi and verifying that the $A/\sqrt{K}$ calculated from the measured leakage is within the specified limit of 10% of 0.054 square feet. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18-month test schedule may be resumed.

* Includes each vacuum relief valve and associated piping.

CONTAINMENT SYSTEMS

DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL AND DRYWELL SPRAY

LIMITING CONDITIONS FOR OPERATION

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

a. One OPERABLE RHR pump, and
b. An OPERABLE flow path capable of recirculating water from the suppression pool through an RHR heat exchanger and the suppression chamber and drywell spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression chamber and/or drywell spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

b. By verifying that each of the required RHR pumps develops a flow of at least 450 gpm on recirculation flow through the RHR heat exchanger and suppression pool spray sparger when tested pursuant to Specification 4.0.5.

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

* 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.

CONTAINMENT SYSTEMS

DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITIONS FOR OPERATION

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

a. One OPERABLE RHR pump, and
b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. By verifying that each of the required RHR pumps develops a flow of at least 7450 gpm on recirculation flow through the RHR heat exchanger and the suppression pool when tested pursuant to Specification 4.0.5.

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

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

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

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

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

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

   4. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with ACTION a.2 or a.3 above, and provided that the associated system is declared inoperable, if applicable, and the appropriate ACTION statements for that system are performed.

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

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

   1. The inoperable valve is returned to OPERABLE status, or
   2. The instrument line is isolated and the associated instrument is declared inoperable.

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

b. At least once per 18 months by removing at least one explosive squib from at least one explosive valve , such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
A. Automatic				
2MSS*AOV6 A,B,C,D	Inside MSIV	1	Z,X,C,D,E,P,T,R,RM,AA	3 to 5
2MSS*AOV7 A,B,C,D	Outside MSIV	1	Z,X,C,D,E,P,T,R,RM,AA	3 to 5
2MSS*MOV208	MSL Drain Line Outside IV	1	Z,X,C,D,E,P,T,R,RM,AA	18
2MSS*MOV111	Main Steam Drain Line Inside IV	1	Z,X,C,D,E,P,T,R,RM,AA	60
2MSS*MOV112	Main Steam Drain Line Outside IV	1	Z,X,C,D,E,P,T,R,RM,AA	60
2RHS*MOV33 A,B	RHS Cont. Spray Outside IVs	*	RM and *	35
2RHS*MOV104	RHS Reactor Head Spray Outside IV	5	A,L,M,Z,RM,CC,DD	50
2RHS*MOV40 A,B	Shutdown Cooling Return Outside IVs	5	A,L,M,Z,RM,CC,DD	29
2RHS*MOV67 A,B	SDC Inboard IV Bypass Valves	5	A,L,M,Z,RM,CC,DD	18
2RHS*MOV112	SDC Supply Inside IV	5	A,L,M,Z,RM,CC,DD	29
2RHS*MOV113	SDC Supply Outside IV	5	A,L,M,Z,RM,CC,DD	29
2CSH*MOV111	CSH Test Return to Suppression Pool Outside IV	*	RM and *	60
2ICS*MOV164	RCIC Vacuum Breaker Outside IV	11	H & F, RM	18
2CCP*MOV94 A,B	CCP Supply to RCS Inside IVs	8	B,F,Z,RM	30
2CCP*MOV17 A,B	CCP Supply to RCS Outside IVs	8	B,F,Z,RM	30
2CCP*MOV16 A,B	CCP Return from RCS Pumps Inside IVs	8	B,F,Z,RM	30
2CCP*MOV15 A,B	CCP Return from RCS Pumps Outside IVs	8	B,F,Z,RM	30
2DFR*MOV120	DFR Drain Tank Line Outside IV	8	B,F,Z,RM	45
2DFR*MOV121	DFR Drain Tank Line Inside IV	8	B,F,Z,RM	45

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2DER*MOV119	DER Line from Drywell Inside IV	8	B,F,Z,RM	35
2DER*MOV120	DER Line from Drywell Outside IV	8	B,F,Z,RM	35
2RCS*SOV104	RCS Sample Inside IV	2	B,C,Z,RM	5
2RCS*SOV105	RCS Sample Outside IV	2	B,C,Z,RM	5
2FPW*SOV218(i)	RCS A Water Spray Outside IV	8	B,F,Z,RM	NA
2FPW*SOV219(i)	RCS A Water Spray Inside IV	8	B,F,Z,RM	NA
2FPW*SOV220(i)	RCS B Water Spray Outside IV	8	B,F,Z,RM	NA
2FPW*SOV221(i)	RCS B Water Spray Inside IV	8	B,F,Z,RM	NA
2DFR*MOV139	DFR Vent Line Outside IV	8	B,F,Z,RM	20
2DFR*MOV140	DFR Vent Line Inside IV	8	B,F,Z,RM	20
2DER*MOV130	DER Vent Line Inside IV	8	B,F,Z,RM	18
2DER*MOV131	DER Vent Line Outside IV	8	B,F,Z,RM	18
2CCP*MOV265	Sply to Drywell Space Cooler Outside IV	8	B,F,Z,RM	60
2CCP*MOV273	Sply to Drywell Space Cooler Inside IV	8	B,F,Z,RM	60
2CCP*MOV122	Return from Drywell Space Cooler Inside IV	8	B,F,Z,RM	60
2CCP*MOV124	Return from Drywell Space Cooler Outside IV	8	B,F,Z,RM	60
2CPS*AOV104	Purge Inlet to Drywell Outside IV	9	B,F,Y,Z,RM	5
2CPS*AOV105	Purge Inlet to Sup. Chamber Outside IV	9	B,F,Y,Z,RM	5
2CPS*AOV106(n)	Purge Inlet to Drywell Inside IV	9	B,F,Y,Z,RM	5
2CPS*AOV107(n)	Purge Inlet to Sup. Chamber Inside IV	9	B,F,Y,Z,RM	5
2CPS*AOV108(n)	Purge Exhaust from Drywell Inside IV	9	B,F,Y,Z,RM	5
2CPS*AOV109(n)	Purge Exhaust from Sup. Chamber Inside IV	9	B,F,Y,Z,RM	5
2CPS*AOV110	Purge Exhaust from Drywell Outside IV	9	B,F,Y,Z,RM	5
2CPS*AOV111	Purge Exhaust from Sup. Chamber Outside IV	9	B,F,Y,Z,RM	5

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2IAS*SOV164	ADS Hdr A N_2 supply Outside IV	8	B,F,Z,RM	5
2IAS*SOV165	ADS Hdr B N_2 supply Outside IV	8	B,F,Z,RM	5
2IAS*SOV166	IAS to MSS Safety Relief Valve Outside IV	8	B,F,Z,RM	5
2IAS*SOV184	IAS to MSS Safety Relief Valve Inside IV	8	B,F,Z,RM	5
2IAS*SOV168	Inst. Air to Testable Check Outside IV	8	B,F,Z,RM	5
2IAS*SOV180	Inst. Air to Testable Check Inside IV	8	B,F,Z,RM	5
2IAS*SOV167	IAS to Test Ck. & Vac. Bkrs. Outside IV	8	B,F,Z,RM	5
2IAS*SOV185	IAS to Test Ck. & Vac. Bkrs. Inside IV	8	B,F,Z,RM	5
2HCS*MOV1 A,B	H_2 Recombiners Sply to Supp. Chamber Outside IV's	8	B,F,Z,RM	30
2HCS*MOV2 A,B	H_2 Recomb. Ret. from Supp. Chamber Outside IV's	8	B,F,Z,RM	30
2HCS*MOV3 A,B	H_2 Recomb. Return from Drywell Outside IV's	8	B,F,Z,RM	30
2HCS*MOV4 A,B(n)	H_2 Recomb. Suply. to Supp. Chamber Inside IV's	8	B,F,Z,RM	30
2HCS*MOV5 A,B(n)	H_2 Recomb. Ret. from Supp. Chamber Inside IV's	8	B,F,Z,RM	30
2HCS*MOV6 A,B(n)	H_2 Recomb. Ret. from Drywell Inside IV's	8	B,F,Z,RM	30
2CPS*SOV119	Containment Purge to Supp. Chamber Outside IV	9	B,F,Y,Z,RM	5
2CPS*SOV120	Containment Purge to Drywell Outside IV	9	B,F,Y,Z,RM	5
2CPS*SOV121(n)	Containment Purge to Supp. Chamber Inside IV	9	B,F,Y,Z,RM	5
2CPS*SOV122(n)	Containment Purge to Drywell Inside IV	9	B,F,Y,Z,RM	5
2CMS*SOV24 A,B,C,D	CMS from Drywell Inside & Outside IV's	8	B,F,Z,RM	5
2CMS*SOV26 A,B,C,D	CMS from SP Inside & Outside IV's	8	B,F,Z,RM	5
2CMS*SOV32 A,B	CMS to Drywell Outside IV's	8	B,F,Z,RM	5
2CMS*SOV33 A,B(n)	CMS to Drywell Inside IV's	8	B,F,Z,RM	5
2CMS*SOV34 A,B(n)	CMS to SP Inside IV's	8	B,F,Z,RM	5
2CMS*SOV35 A,B	CMS to SP Outside IV's	8	B,F,Z,RM	5
2CMS*SOV60 A,B	CMS to Drywell Outside IV's	8	B,F,Z,RM	5
2CMS*SOV61 A,B(n)	CMS to Drywell Inside IV's	8	B,F,Z,RM	5
2CMS*SOV62 A,B	CMS to Drywell Outside IV's	8	B,F,Z,RM	5
2CMS*SOV63 A,B(n)	CMS to Drywell Inside IV's	8	B,F,Z,RM	5

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2CPS*SOV132	Nitrogen to 2CPS*AOV107 Outside IV	9	B,F,Y,Z,RM	5
2CPS*SOV133	Nitrogen to 2CPS*AOV109 Outside IV	9	B,F,Y,Z,RM	5
2LMS*SOV152(i)	LMS from Drywell Inside IV	8	B,F,Z,RM	5
2LMS*SOV153(i)	LMS from Drywell Outside IV	8	B,F,Z,RM	5
2LMS*SOV156(i)	LMS from SP Inside IV	8	B,F,Z,RM	5
2LMS*SOV157(i)	LMS from SP Outside IV	8	B,F,Z,RM	5
2RCS*SOV65 A,B(l)	Hyd. Unit to RCS FCVs Outside IV's	8	B,F,Z,RM	20
2RCS*SOV66 A,B(l)	Hyd. Unit to RCS FCVs Outside IV's	8	B,F,Z,RM	20
2RCS*SOV67 A,B(l)	Hyd. Unit to RCS FCVs Outside IV's	8	B,F,Z,RM	20
2RCS*SOV68 A,B(l)	Hyd. Unit from RCS FCVs Outside IV's	8	B,F,Z,RM	20
2RCS*SOV79 A,B(l)	Hyd. Unit to RCS FCVs Inside IV's	8	B,F,Z,RM	20
2RCS*SOV80 A,B(l)	Hyd. Unit to RCS FCVs Inside IV's	8	B,F,Z,RM	20
2RCS*SOV81 A,B(l)	Hyd. Unit to RCS FCVs Inside IV's	8	B,F,Z,RM	20
2RCS*SOV82 A,B(l)	Hyd. Unit from RCS FCVs Inside IV's	8	B,F,Z,RM	20
2ICS*MOV121	RCIC Steam Supply Outside IV	10	K,M,H,Z,RM,BB,CC,DD	14
2ICS*MOV128(n)	RCIC Steam Supply Inside IV	10	K,M,H,RM,BB,CC,DD	14
2ICS*MOV170	RCIC Warmup Valve Inside IV	10	K,M,H,RM,BB,CC,DD	10
2WCS*MOV102	WCS Supply from RCS & RPV Inside IV	7	B,J,U,S,Z,RM,DD	14
2WCS*MOV112	WCS Supply from RCS & RPV Outside IV	6	B,J,U,S,Z,RM,DD	14
2ICS*MOV148	RCIC Vacuum Breaker Outside IV	11	H & F, RM	18
2NMS*SOV1 A, B, C, D, E	Traversing Incore Probe Ball Outside IV's	3	B,F,Z,RM	5
2GSN*SOV166	Nitrogen Purge to TIP Indexing Mechanism Outside IV	3	B,F,Z,RM	5

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2RHS*MOV142(j)(m)	RHS Drain to Radwaste Outside IV	4	A,Z,F,RM	30
2RHS*MOV149(j)(m)	RHS Drain to Radwaste Inside IV	4	A,Z,F,RM	25
2RHS*SOV35 A/B (j)(m)	RHS Sample HX Inside IVs	4	A,Z,F,RM	5
2RHS*SOV36 A/B (j)(m)	RHS Sample HX Outside IVs	4	A,Z,F,RM	5
2RDS*AOV124(k)	SCRAM Discharge volume vent	NA		NA
2RDS*AOV132(k)	SCRAM Discharge volume vent	NA		NA
2RDS*AOV123(k)	SCRAM Discharge volume drain	NA		NA
2RDS*AOV130(k)	SCRAM Discharge volume drain	NA		NA
B. Remote Manual				
2RHS*MOV15 A,B	Containment Spray to Drywell Outside IV's	12	RM	NA
2RHS*MOV 1 A,B,C	RHS Pump Suction Outside IVs	12	RM	NA
2RHS*MOV30 A,B	RHS Test Line to SP Outside IVs	12	RM	NA
2RHS*MOV25 A,B(n)	Containment Spray to Drywell Outside IVs	12	RM	NA
2RHS*MOV24 A,B,C	RHS/LPCI to RPV Outside IVs	12	RM	NA
2CSH*MOV118(n)	CSH Suction from SP Outside IV	12	RM	NA
2CSH*MOV105	HPCS Min Flow Bypass Outside IV	12	RM	NA
2CSH*MOV107	CSH to RPV Outside IV	12	RM	NA
2CSL*MOV112	CSL Suction from SP Outside IV	12	RM	NA
2CSL*MOV104	CSL to RPV Outside IV	12	RM	NA
2ICS*MOV136(n)	ICS Suction from SP Outside IV	12	RM	NA
2ICS*MOV143(n)	ICS Min flow to SP Outside IV	12	RM	NA

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2ICS*MOV122(n)	ICS turbine exhaust to SP Outside IV	12	RM	NA
2ICS*MOV126	ICS to RPV Outside IV	12	RM	NA
2NMS*VEX1 A, B, C, D, E(d)	Traversing Incore Probe Shear Outside IVs	12	RM	NA
2FWS*MOV21 A,B	Feedwater to RPV Outside IVs	12	RM	NA
2WCS*MOV200	WCS to RPV Outside IV	12	RM	NA
2RHS*MOV26 A,B(c)	RHS HX vent Inboard IVs	12	RM	NA
2RHS*MOV27 A,B(c)	RHS HX vent Outboard IVs	12	RM	NA
2MSS*SOV97 A,B,C, D(n)	Main Steam Line Drains	12	RM	NA
2SLS*MOV5 A,B(g)	SLS to RPV Outside IV	12	RM	NA
C. Manual				
2SAS*HCV160	SAS to Drywell Outside IV			
2SAS*HCV161	SAS to Drywell Outside IV			
2SAS*HCV162	SAS to Drywell Inside IV			
2SAS*HCV163	SAS to Drywell Inside IV			
2AAS*HCV134	AAS to Drywell Outside IV			
2AAS*HCV135	AAS to Drywell Outside IV			
2AAS*HCV136	AAS to Drywell Inside IV			
2AAS*HCV137	AAS to Drywell Inside IV			
2RHS*V192	RCIC/RHS Vacuum Breaker Outside IV			
2SFC*V203	Inner Refuel Seal Leakoff Outboard IV			
2SFC*V204	Inner Refuel Seal Leakoff Inboard IV			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
D. Other				
Safety Relief				
2RHS*RV20 A,B,C(d)	RHS RV disch. to SP Outside IVs			
2RHS*RV61 A,B,C(d)	RHS RV disch. to SP Outside IVs			
2RHS*RV108(d)	RHS RV disch. to SP Outside IVs			
2RHS*RV110(d)	SDC to RHS Pump suction RV			
2RHS*RV139(d)	RHR Hdr. Flush to Radwaste RV			
2RHS*RV152(n)	SDC Supply from RCS RV Inside IV			
2RHS*RV56 A,B(d)	RHS HX shell side RVs			
2RHS*SV34 A,B(d)	RHS HX steam supply Safety valves			
2RHS*SV62 A,B(d)	RHS HX steam supply Safety valves			
2RHS*RVV35 A,B(d)	RHS Vacuum Breakers			
2CSL*RV105(d)	CSL RV Disch. to SP Outside IV			
2CSL*RV123(d)	CSL RV Disch. to SP Outside IV			
2RHS*RVV36 A,B(d)	RHS Vacuum Breakers			
2CCP*RV170(n)	CCP RV Discharge Inside IV			
2CCP*RV171(n)	CCP RV Discharge Inside IV			
2CSH*RV113(d)	CSH RV Disch. to SP Outside IV			
2CSH*RV114(d)	CSH RV Disch. to SP Outside IV			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
Check Valves				
2RHS*AOV16 A,B,C(h)	RHS/LPCI to RPV Inside IVs			
2RHS*AOV39 A,B(h)	SDC to RCS Inside IVs			
2CPS*V50	Nitrogen Supply to 2CPS*AOV107 Inside IV			
2CPS*V51	Nitrogen Supply to 2CPS*AOV109 Inside IV			
2CSH*AOV108(h)	CSH to RPV Inside IV			
2CSL*AOV101(h)	CSL to RPV Inside IV			
2ICS*AOV156(h)	ICS to RPV Outside IV			
2ICS*AOV157(h)	ICS to RPV Inside IV			
2SLS*V10	SLS to RPV Inside IV			
2GSN*V170	N_2 Purge to Tip Index Mech. Inside IV			
2IAS*V448	IAS to ADS Accumulators Inside IV			
2IAS*V449	IAS to ADS Accumulators Inside IV			
2RCS*V59 A,B	RDS to RCS Pump A Seal Outside IVs			
2RCS*V60 A,B	RDS to RCS Pump A Seal Inside IVs			
2RCS*V90 A,B	RDS to RCS Pump A Seal Outside IVs			
2RHS*V19(d)(f)	Discharge Check from RCIC to Supp. Pool			
2RHS*V20(d)(f)	Discharge Check from RCIC to Supp. Pool			
2RHS*V117(d)(f)	Check Valve from RCIC Drain to Supp. Pool			
2RHS*V118(d)(f)	Check Valve from RCIC Drain to Supp. Pool			
2FWS*AOV23 A,B(h)	Feedwater to RPV Outside IV's			
2FWS*V12 A,B	Feedwater to RPV Inside IV's			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
Excess Flow Check(e) Reactor Instrumentation Lines				
2ISC*EFV1	Inst. Line from MSS			
2ISC*EFV2	Inst. Line from N14,200°			
2ISC*EFV3	Inst. Line from N14,160°			
2ISC*EFV4	Inst. Line from N13,190°			
2ISC*EFV5	Inst. Line from N14,20°			
2ISC*EFV6	Inst. Line from N14,340°			
2ISC*EFV7	Inst. Line from N13,10°			
2ISC*EFV8	Inst. Line from N12,160°			
2ISC*EFV10	Inst. Line from N12,200°			
2ISC*EFV11	To 2ISC*FT47K,FT48B			
2ISC*EFV13	To 2ISC*FT47H			
2ISC*EFV14	Vessel Bottom Tap, loop A Jet Pump			
2ISC*EFV15	Inst. Line from N12,340°			
2ISC*EFV17	Inst. Line from N12,20°			
2ISC*EFV18	To 2ISC*FT47J,FT48A			
2ISC*EFV20	To 2ISC*FT47E			
2ISC*EFV21	Vessel Bottom Tap for CSH, RDS			
2ISC*EFV22	Vessel Bottom Tap for WCS and Loop B J.P.			
2ISC*EFV23	To 2ISC*FT48C and Postaccident Sampling			
2ISC*EFV24	To 2ISC*FT48D and Postaccident Sampling			
2ISC*EFV25	To 2ISC*FT47L			
2ISC*EFV26	To 2ISC*FT47C			
2ISC*EFV27	To 2ISC*FT47A			
2ISC*EFV28	To 2ISC*FT47R			
2ISC*EFV29	To 2ISC*FT47G			
2ISC*EFV30	To 2ISC*FT47N			
2ISC*EFV31	To 2ISC*FT48A			
2ISC*EFV32	To 2ISC*FT47T			
2ISC*EFV33	To 2ISC*FT47V,FT48C			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2ISC*EFV34	To 2ISC*FT47B			
2ISC*EFV35	To 2ISC*FT47D			
2ISC*EFV36	To 2ISC*FT47F			
2ISC*EFV37	To 2ISC*FT47S			
2ISC*EFV38	To 2ISC*FT47M			
2ISC*EFV39	To 2ISC*FT47P			
2ISC*EFV40	To 2ISC*FT48B			
2ISC*EFV41	To 2ISC*FT47U			
2ISC*EFV42	To 2ISC*FT47W,FT48D			
2ISC*EFV9	Containment Pressure 2ISC*PT15C, 16B, 16D			
2ISC*EFV12	Containment Pressure 2ISC*PT15B, 17B, 17D			
2ISC*EFV16	Containment Pressure 2ISC*PT15A, 16A, 16C			
2ISC*EFV19	Containment Pressure 2ISC*PT15D, 17A, 17C			
2CMS*EFV1A	To CMS*PT1A			
2CMS*EFV1B	To CMS*PT1B			
2CMS*EFV3A	To CMS*PT2A			
2CMS*EFV3B	To CMS*PT2B			
2CMS*EFV5A	To CMS*PT7A			
2CMS*EFV5B	To CMS*PT7B			
2CMS*EFV6	To CMS-PT168			
2CMS*EFV8A	To CMS*LT9A, 11A, 114			
2CMS*EFV8B	To CMS*LT9B, 11B, 105			
2CMS*EFV9A	To CMS*LT9A, 11A, 114			
2CMS*EFV9B	To CMS*LT9B, 11B, 105			
2CMS*EFV10	To CMS-PI173			
2ICS*EFV1	To 2ICS*PDT167			
2ICS*EFV2	To 2ICS*PDT167			
2DER*EFV31	To DER*PT134			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2ICS*EFV3	To 2ICS*PDT168			
2ICS*EFV4	To 2ICS*PDT168			
2IAS*EFV200	To 2IAS*PT230 off ADS Accum.			
2IAS*EFV201	To 2IAS*PT231 off ADS Accum.			
2IAS*EFV202	To 2IAS*PT232 off ADS Accum.			
2IAS*EFV203	To 2IAS*PT233 off ADS Accum.			
2IAS*EFV204	To 2IAS*PT234 off ADS Accum.			
2IAS*EFV205	To 2IAS*PT235 off ADS Accum.			
2IAS*EFV206	To 2IAS*PT236 off ADS Accum.			
2RHS*EFV 5, 6	To 2RHS*PDT18B			
2RHS*EFV7	To 2RHS*PDT18A			
2MSS*EFV 1A,B,C,D	To Flow elements A,B,C,D steamlines			
2MSS*EFV 2A,B,C,D	To Flow elements A,B,C,D steamlines			
2MSS*EFV 3A,B,C,D	To Flow elements A,B,C,D steamlines			
2MSS*EFV 4A,B,C,D	To Flow elements A,B,C,D steamlines			
2RCS*EFV44 A,B	To 2RCS*PT 84 A/B			
2RCS*EFV45 A,B	To 2RCS*FT 7 A/B, FT 9 A/B			
2RCS*EFV46 A,B	To 2RCS*FT 7 A/B, FT 9 A/B			
2RCS*EFV47 A,B	To 2RCS*FT 6 A/B, FT 8 A/B			
2RCS*EFV48 A,B	To 2RCS*FT 6 A/B, FT 8 A/B			
2RCS*EFV52 A,B	To 2RCS*PDT 15 A/B			
2RCS*EFV53 A,B	To 2RCS*PDT 15 A/B			
2RCS*EFV62 A,B	To 2RCS*PT44 A/B			
2RCS*EFV63 A,B	To 2RCS*PT42 A/B			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

ISOLATION VALVE NO.	VALVE FUNCTION	VALVE GROUP	ISOLATION SIGNAL(a)	MAXIMUM CLOSING TIME (SECONDS)
2WCS*EFV221	To 2WCS-FT 134			
2WCS*EFV222	To 2WCS*FT67X, PDS 115			
2WCS*EFV223	To 2WCS*FT67Y			
2WCS*EFV224	To 2WCS*FT67Y			
2WCS*EFV300	To 2WCS*FT67X, PDS 115			
2CSH*EFV1	To 2CSH*LT123, LT124			
2CSH*EFV2	To 2CSH*LT123, LT124			
2CSH*EFV3	To 2CSH*PDT109			
2CSL*EFV1	To 2CSL*PDT132 and 2RHS*PDT18A			

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATION

* Isolates on injection signal, not primary containment isolation signal.

(a) See Specification 3.3.2, Table 3.3.2-4, for valve groups operated by isolation signal(s).

(b) Deleted.

(c) These valves are the RHR heat exchangers vent lines isolation valves. The vent line connects to the RHR safety relief valves (SRVs) Discharge Header before it penetrates the primary containment. The position indicators for these valves are provided in the Control Room for remote manual isolation.

(d) Type C leakage tests not required.

(e) The associated instrument lines shall not be isolated during Type A testing. Type C testing is not required. These valves shall be tested in accordance with Surveillance Requirement 4.6.3.4.

(f) These valves are check valves, located on the vacuum breaker lines for RHR SRVs discharge headers. The SRV discharge header terminates under pool water and therefore has no containment isolation valves other than those on lines feeding into it.

(g) 2SLS*MOV5A and B are globe stop check valves. These valves close upon reverse flow. The motor operator is provided to remote manually close the valve from the control room.

(h) These valves are testable check valves. They close upon reverse flow. The air operator on each valve is provided only for periodic testing of the valve. These valves can only be tested against a zero d/p.

(i) Valves are maintained closed. The FPW lines are capped. Valves are Type C tested.

(j) Not primary containment penetration isolation valves. These valves close on an isolation signal to provide integrity of "A" and "B" LPCI loops.

(k) Valves close on a SCRAM signal; not part of primary containment isolation system but are included here for Type C testing per Specification 3.6.1.2. These valves are not required to be OPERABLE per this specification but are required to be OPERABLE per Specification 3.1.3.1.

(l) Not subject to Type A or Type C leak test because of constant monitoring under constant 1800 psig pressure and the possible detrimental effects of shutdown.

(m) Not subject to Type C test per 10 CFR 50, Appendix J. A hydrostatic test is performed in accordance with Specification 4.6.1.2.d.3.

(n) These valves are Type C tested and may be tested in the reverse direction.

3/4.6.4 SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS

LIMITING CONDITIONS FOR OPERATION

3.6.4 All suppression chamber/drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more vacuum breakers in one pair of suppression chamber/drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pair of vacuum breakers 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 suppression chamber/drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the position indicator of any suppression chamber/drywell vacuum breaker inoperable verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 24 hours thereafter; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

CONTAINMENT SYSTEMS

SUPPRESSION CHAMBER/DRYWELL VACUUM BREAKERS

SURVEILLANCE REQUIREMENTS

4.6.4 Each suppression chamber/drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.
b. Demonstrated OPERABLE:

   1. At least once per 31 days and within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

   2. At least once per 31 days by verifying the position indicator(s) OPERABLE by observing expected valve movement during the cycling test.*

   3. At least once per 18 months by;

      a) Verifying the opening setpoint, from the closed position, to be less than or equal to 0.25 psid, and

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

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* Observation of expected valve movement during cycling test will be accomplished for the purposes of this surveillance by observing valve position indicators in the control room.

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. All secondary containment equipment hatches are closed and sealed.

   2. At least one door in each access to the secondary containment is closed, except during normal entry and exit.

   3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

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* When irradiated fuel is being handled in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.6.5.1 (Continued)

c. At least once per 18 months:

1. Verifying that each standby gas treatment subsystem will draw down the secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 120 seconds when starting at a pressure no less than zero psig, and

2. Operating one standby gas treatment subsystem for 1 hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the secondary containment at a flow rate not exceeding 3190 cfm.

SECONDARY CONTAINMENT

SECONDARY CONTAINMENT AUTOMATIC ISOLATION DAMPERS

LIMITING CONDITIONS FOR OPERATION

3.6.5.2 The secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

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

b. Isolate each affected penetration by use of at least one deactivated damper secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual dampers 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

4.6.5.2 Each secondary containment ventilation system automatic isolation damper shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Before returning the damper to service after maintenance, repair, or replacement work is performed on the damper or on 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

4.6.5.2 (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 at least once per 92 days.

TABLE 3.6.5.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS

DAMPER FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)
Reactor Building Ventilation Supply Damper 2HVR*AOD1A and 2HVR*AOD1B	6
Reactor Building Ventilation Exhaust Damper 2HVR*AOD9A and 2HVR*AOD9B	5
Reactor Building Ventilation Exhaust Damper 2HVR*AOD10A and 2HVR*AOD10B	5

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.6.5.3 Two independent standby gas treatment (SGTS) subsystems shall be OPERABLE.

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

ACTION:

a. With one standby gas treatment subsystem inoperable:

   1. In OPERATIONAL CONDITION 1, 2 or 3, suspend all VENTING or PURGING of the drywell and/or suppression chamber** within 30 minutes, and 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.

   2. In OPERATIONAL CONDITION *, restore the inoperable subsystem to OPERABLE status within 7 days, or suspend handling of irradiated fuel in the reactor building, 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:

   1. In OPERATIONAL CONDITION 1, 2, or 3, suspend all operations involving VENTING, PURGING, or pressure control of the drywell or suppression chamber and initiate action within 1 hour to 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 reactor building, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

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* When irradiated fuel is being handled in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** The requirement to suspend VENTING or PURGING with one inoperable SGTS subsystem shall not apply to the use of valves 2CPS*AOV108 (14-inch) and 2CPS*AOV110 (14-inch), or 2CPS*AOV109 (12-inch) and 2CPS*AOV111 (12-inch), for primary containment pressure control, provided 2GTS*AOV101 is closed, and its 2-inch bypass line is the only flow path to the standby gas treatment system.

STANDBY GAS TREATMENT SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters OPERABLE.

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 Positions C.5.a, C.5.c, and C.5.d of RG 1.52*, Revision 2, March 1978, and the subsystem 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 Position C.6.b of RG 1.52*, Revision 2, March 1978, meets the laboratory testing criteria of Position C.6.a of RG 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 system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Position C.6.b of RG 1.52*, Revision 2, March 1978, meets the laboratory testing criteria of Position C.6.a of RG 1.52*, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%.

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* ANSI N510-1980 is applicable in place of ANSI N510-1975, and ANSI N509-1980 is applicable in place of ANSI N509-1976.

CONTAINMENT SYSTEMS

SECONDARY CONTAINMENT

STANDBY GAS TREATMENT SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.5.3 (Continued)

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.5 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm ± 10%.

   2. Verifying that the filter train starts and isolation valves open on each of the following test signals:

      a. Manual initiation from the control room, and
      b. Simulated automatic initiation signal.

   3. Verifying that the decay heat removal air inlet valves are closed and can be manually opened.

   4. Verifying that the heaters dissipate 20.0 ± 2.0 kW when tested in accordance with ANSI N510-1980.

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 N510-1980 while operating the system at a flow rate of 4000 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 N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ± 10%.

CONTAINMENT SYSTEMS

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater coil outlet gas temperature increases to greater than or equal to 1150°F within 90 minutes. Maintain 1150°F or more for at least 4 hours.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner 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 1 million ohms.

c. By measuring the system leakage rate:

   1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

   2. By measuring the leakage rate of the system outside of the containment isolation valves at Pa, 39.75 psig, on the schedule required by Specification 4.6.1.2, and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITIONS FOR OPERATION

3.6.6.2 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

a. Within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to

b. Within 24 hours before reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

With the oxygen concentration in the drywell and/or suppression chamber exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.2 The oxygen concentration in the drywell and suppression chamber shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and at least once per 7 days thereafter.

* See Special Test Exception 3.10.5.

3/4.7 PLANT SYSTEMS

3/4.7.1 PLANT SERVICE WATER SYSTEM

PLANT SERVICE WATER SYSTEM - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.7.1.1 Two independent plant service water system loops shall be OPERABLE with one loop in operation. Each loop shall be comprised of:

a. Two plant service water pumps capable of taking suction from Lake Ontario and transferring the water to the associated safety-related equipment.

b. Service water supply header discharge water temperature of 76°F or less.

The intake deicing heater system shall be OPERABLE and in operation when intake tunnel water temperature is less than 39°F; Division I shall have 7 heaters in operation in each intake structure and Division II shall have 7 heaters in operation in each intake structure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one less than the required number of OPERABLE plant service water pumps in one loop, restore the inoperable pump 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 one less than the required number of OPERABLE plant service water pumps in each loop, restore at least one inoperable pump to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With two less than the required number of OPERABLE plant service water pumps in one loop or with one plant service water loop otherwise inoperable, restore at least one pump 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.

d. With two less than the required number of OPERABLE plant service water pumps in one loop and one less than the required number of plant service water pumps in the other loop, restore at least one of the two inoperable pumps in the same loop 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.

e. With two plant service water system loops OPERABLE and the service water supply header discharge water temperature continuously exceeding 76°F for any 8 hour period, within one hour initiate action to be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

PLANT SERVICE WATER SYSTEM - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.7.1.1 (Continued)

ACTION:

f. With less than the required Division I and Division II heaters OPERABLE within one hour initiate action to be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1.1 The plant service water system shall be demonstrated OPERABLE:

a. By verifying the plant service water supply header discharge water temperature to be less than or equal to 76°F:

   1. At least once per 24 hours, and

   2. At least once per 4 hours when the last recorded water temperature is greater than or equal to 70°F, and

   3. At least once per 2 hours when the last recorded water temperature is greater than or equal to 74°F.

b. At least once per 12 hours by verifying the water level at the service water pump intake is greater than or equal to elevation 233.1 feet.

c. At least once per 31 days by verifying that each valve - manual, power-operated, or automatic, servicing safety-related equipment that is not locked, sealed, or otherwise secured in position - is in its correct position.

d. At least once per 18 months during shutdown, by verifying:

   1. After a simulated test signal, each automatic valve servicing non-safety-related equipment actuates to its isolation position.

   2. After a simulated test signal, each service water system cross connect and pump discharge valve actuates automatically to its isolation position.

   3. For each service water pump, after a simulated test signal, the pump starts automatically and the associated pump discharge valve opens automatically, in order to supply flow to the system safety-related components.

SURVEILLANCE REQUIREMENTS

4.7.1.1.1.d (Continued)

4. Each pump runs and maintains service water pump discharge pressure equal to or greater than 80 psig with a pump flow equal to or greater than 6500 gpm.

5. The resistance is ≥ 28 ohms for each feeder cable and associated heater elements in the intake deicing heater systems.

e. At least once per 18 months, perform a LOGIC SYSTEM FUNCTIONAL TEST of the service water pump starting logic.

4.7.1.1.2 The Intake Deicing Heater System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the intake tunnel water temperature is greater than or equal to 39°F, or

b. At least once per 7 days by verifying that the current of the heater feeder cables at the motor control centers is 10 amps* or more (total for three phases) at ≥ 518 volts per divisional heater in each intake structure.

* For 7 heater elements in operation.

PLANT SYSTEMS

PLANT SERVICE WATER SYSTEM

PLANT SERVICE WATER SYSTEM - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.7.1.2 Two independent plant service water system loops shall be OPERABLE with one loop in operation. Each loop shall be comprised of:

a. Two OPERABLE plant service water pumps capable of taking suction from Lake Ontario and transferring the water to the associated safety-related equipment.

b. Service water supply header discharge water temperature of 76°F or less.

The intake deicing heater systems shall be OPERABLE and in operation when intake tunnel water temperature is less than 39°F; Division I shall have 7 heaters in operation in each intake structure and Division II shall have 7 heaters in operation in each intake structure.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

a. With one less than the required number of OPERABLE plant service water pumps in one loop, restore the inoperable pump to OPERABLE status within 30 days or declare the associated safety-related equipment inoperable and take ACTIONS required by Specifications 3.5.2 and 3.8.1.2.

b. With one less than the required number of OPERABLE plant service water pumps in each loop, restore at least one inoperable pump to OPERABLE status within 7 days or declare the associated safety-related equipment inoperable and take ACTIONS required by Specification 3.5.2 and 3.8.1.2.

c. With two less than the required number of OPERABLE plant service water pumps in one loop, restore at least one inoperable pump to OPERABLE status within 72 hours or declare the associated safety-related equipment inoperable and take ACTIONS required by Specification 3.5.2 and 3.8.1.2.

d. With two less than the required number of OPERABLE plant service water pumps in one loop and one less than the required number of plant service water pumps in the other loop, restore at least one of the two inoperable pumps in the same loop to OPERABLE status within 12 hours or declare the associated safety-related equipment inoperable and take ACTIONS required by Specification 3.5.2 and 3.8.1.2.

e. With the service water supply header discharge temperature exceeding 76°F, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel.

LIMITING CONDITIONS FOR OPERATION

3.7.1.2 (Continued)

ACTION:

f. With less than the required Division I and Division II heaters OPERABLE, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The plant service water system shall be demonstrated OPERABLE:

a. By verifying the plant service water supply header discharge water temperature to be less than or equal to 76°F:

   1. At least once per 24 hours, and

   2. At least once per 4 hours when the last recorded water temperature is greater than or equal to 70°F, and

   3. At least once per 2 hours when the last recorded water temperature is greater than or equal to 74°F.

b. At least once per 12 hours by verifying the water level at the service water pump intake is greater than or equal to elevation 233.1 feet.

c. At least once per 31 days by verifying that each valve - manual, power-operated, or automatic, servicing safety-related equipment that is not locked, sealed, or otherwise secured in position - is in its correct position.

d. At least once per 18 months during shutdown, by verifying:

   1. After a simulated test signal, each automatic valve servicing non-safety-related equipment actuates to its isolation position.

   2. After a simulated test signal, each service water system cross connect and pump discharge valve actuates automatically to its isolation position, and

   3. For each service water pump, after a simulated test signal, the pump starts automatically and the associated pump discharge valve opens automatically, in order to supply flow to the system safety-related components.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1.d (Continued)

4. Each pump runs and maintains service water pump discharge pressure equal to or greater than 80 psig with each pump flow equal to or greater than 6500 gpm.

5. The resistance is $\geq$ 28 ohms for each feeder cable and associated heater elements in the intake deicing heater systems.

e. At least once per 18 months, perform a LOGIC SYSTEM FUNCTIONAL TEST of the service water pump starting logic.

4.7.1.2.2 The Intake Deicing Heater System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying the intake tunnel water temperature is greater than or equal to 39°F, or

b. At least once per 7 days by verifying that the current of the heater feeder cables at the motor control centers is 10 amps* or more (total for 3 phases) at $\geq$ 518 volts per divisional heater in each intake structure.

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* For 7 heater elements in operation.

3/4.7.2 REVETMENT-DITCH STRUCTURE

LIMITING CONDITIONS FOR OPERATION

3.7.2 The revetment-ditch structure shall be structurally sound and capable of limiting wave action as intended. The revetment-ditch structure shall be maintained so that the elevation of each survey point listed in Table 3.7.2-1 is not more than 1.0 foot below the listed elevation.

APPLICABILITY: At all times.

ACTION:

With the elevation of one or more survey points more than 1 foot below the elevation given in Table 3.7.2-1, prepare and submit to the Commission within 90 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

a. Explanation of how the elevation change occurred and if the revetment-ditch structure is continuing to change;

b. A planned course of repair (if required) and a schedule for accomplishing the repair;

c. Evaluation of and justification for continued plant operation; and

d. The current elevation of each survey point shown in Table 3.7.2-1.

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

REVETMENT-DITCH STRUCTURE

SURVEILLANCE REQUIREMENTS

4.7.2 The revetment-ditch structure shall be capable of limiting wave action and shall be determined to be structurally sound by performing:

a. A visual and a limited survey* inspection once a year.

b. A full survey* before initial fuel load and once every 2 years thereafter, provided that the maximum elevation change of one or more control points in that period is less than 2 inches. If the elevation change of one or more control points exceeds 2 inches in any 2-year period, a full survey will be performed once a year thereafter, or until two consecutive surveys shows any additional elevation change to be less than 2 inches in each 2-year period;

c. A visual inspection within 7 days and a full survey* as soon as practical after any earthquake event with an intensity greater than the operating basis earthquake (OBE).

*Limited and full surveys shall be performed with survey equipment to at least third-order accuracy.

TABLE 3.7.2-1

SURVEY POINTS FOR REVETMENT-DITCH STRUCTURE

SURVEY POINT*,**	NORTH-SOUTH	EAST-WEST	OCTOBER 1985 CONTROL ELEVATION
1C	N 1283798.13	E 546895.34	252.165
1B	N 1283821.89	E 546888.75	263.075
1A	N 1283828.48	E 546887.88	264.290
2C	N 1283770.08	E 546799.33	253.765
2B	N 1283793.42	E 546793.66	263.725
2A	N 1283800.75	E 546802.21	264.535
3C	N 1283741.94	E 546703.71	252.920
3B	N 1283766.13	E 546696.05	263.130
3A	N 1283771.53	E 546695.04	263.215
4C	N 1283660.46	E 546608.29	254.450
4B	N 1283681.23	E 546606.66	263.560
4A	N 1283684.30	E 546597.26	263.950
5C	N 1283688.87	E 546510.60	255.665
5B	N 1283705.21	E 546496.47	263.820
5A	N 1283714.96	E 546498.16	263.440
6C	N 1283672.49	E 546405.17	255.870
6B	N 1283689.60	E 546407.84	264.180
6A	N 1283694.74	E 546404.07	264.756
7C	N 1283675.92	E 546305.02	256.575
7B	N 1283696.02	E 546299.75	263.680
7A	N 1283705.09	E 546309.27	265.235
8C	N 1283684.19	E 546205.09	256.300
8B	N 1283705.28	E 546205.14	263.125
8A	N 1283712.00	E 546206.38	263.155
9C	N 1283691.57	E 546110.39	257.715
9B	N 1283710.39	E 546108.25	263.660
9A	N 1283720.79	E 546102.81	263.090
10C	N 1283673.00	E 546016.51	257.135
10B	N 1283693.66	E 546004.95	265.540
10A	N 1283701.25	E 546006.68	264.035

* See Bases Figures B3/4 7.2-1 and B3/4 7.2-2 for location sketches.

** Survey points are anchored into back armor using stainless steel HILTI quick bolts.

TABLE 3.7.2-1 (Continued)

SURVEY POINTS FOR REVETMENT-DITCH STRUCTURE

SURVEY POINT*,**	NORTH-SOUTH	EAST-WEST	OCTOBER 1985 CONTROL ELEVATION
11C	N 1283646.76	E 545918.23	256.155
11B	N 1283666.83	E 545918.76	263.720
11A	N 1283675.17	E 545912.02	263.385
12C	N 1283626.37	E 545839.53	256.395
12B	N 1283650.74	E 545835.16	264.085
12A	N 1283656.67	E 545831.43	264.105

* See Bases Figures B3/4 7.2-1 and B3/4 7.2-2 for location sketches.

** Survey points are anchored into back armor using stainless steel HILTI quick bolts.

PLANT SYSTEMS

3/4.7.3 CONTROL ROOM OUTDOOR AIR SPECIAL FILTER TRAIN SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.3 Two independent control room outdoor air special filter trains* shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and **.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with one control room filter train inoperable, restore the inoperable filter train to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 4, 5, or **:

   1. With one control room filter train inoperable, restore the inoperable filter train to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE filter train in the emergency pressurization mode of operation.

   2. With both control room filter trains inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the reactor building and operations with a potential for draining the reactor vessel.

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

SURVEILLANCE REQUIREMENTS

4.7.3 Each control room outdoor air special filter train 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 90°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 filter train operates for at least 10 hours with the heaters OPERABLE.

* This includes the control room chiller subsystem.

** When irradiated fuel is being handled in the reactor building and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel and uncovering irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.3 (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 filter trains by:

   1. Verifying that the filter train satisfies the in-place penetration and bypass testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Positions C.5.a, C.5.c, and C.5.d of RG 1.52*, Revision 2, March 1978, and the system flow rate is 2250 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Position C.6.b of RG 1.52*, Revision 2, March 1978, meets the laboratory testing criteria of Position C.6.a of RG 1.52*, Revision 2, March 1978, for a methyl iodide penetration of less than 0.175%; and

   3. Verifying a subsystem flow rate of 2250 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1980.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Position C.6.b of RG 1.52*, Revision 2, March 1978, meets the laboratory testing criteria of Position C.6.a of RG 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 5.5 inches water gauge (WG) while operating the subsystem at a flow rate of 2250 cfm ± 10%.

   2. Verifying that on each of the below pressurization mode actuation test signals, the filter train automatically switches to the emergency pressurization mode of operation and the control room is maintained at a positive pressure of 1/8 inch WG relative to the outside atmosphere during subsystem operation at an outside air intake flow rate less than or equal to 1500 cfm.

      (a) Air intake radiation monitors, and

      (b) LOCA, and

* ANSI N510-1980 is applicable in place of ANSI N510-1975, and ANSI N509-1980 is applicable in place of ANSI N509-1976.

SURVEILLANCE REQUIREMENTS

4.7.3.e (Continued)

3. Verifying that the heaters dissipate 7.95 kW or more when tested in accordance with ANSI N510-1980.

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 N510-1980 while operating the system at a flow rate of 2250 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 N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2250 cfm ± 10%.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 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 150 psig or less within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

   3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of 600 gpm or more in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

* 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

4.7.4 (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 600 gpm or more 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.

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

3/4.7.5 SNUBBERS

LIMITING CONDITIONS FOR OPERATION

3.7.5 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

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, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.5 on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.5 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each category may be inspected independently according to the schedule below. The first inservice visual inspection of snubbers shall be performed after 2 months but within 12 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers 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:

PLANT SYSTEMS

SNUBBERS

SURVEILLANCE REQUIREMENTS

4.7.5.b (Continued)

NO. INOPERABLE SNUBBERS 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 (1) that there are no visible indications of damage or impaired OPERABILITY, (2) that attachments to the foundation or supporting structure are OPERABLE, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are OPERABLE. Snubbers which appear inoperable as a result of these visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of the 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 Surveillance Requirement 4.7.5.f.

d. Transient Event Inspection

An inspection shall be performed of all 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 shall not be lengthened more than one step at a time.

**The provisions of Specification 4.0.2 are not applicable.

4.7.5 (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 plans for each type of snubber. The sample plan shall be selected before 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 before the test period or the sample plan used in the previous 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 4.7.5.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 4.7.5-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.5.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 4.7.5-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the 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 of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be

4.7.5.e.3 (Continued)

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 have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested. 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 that failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required because of failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression;

2. For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

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

4.7.5.g (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 4.7.5.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 all 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 on the basis of engineering information and shall be extended or shortened on the basis of 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 6.10.1.2.

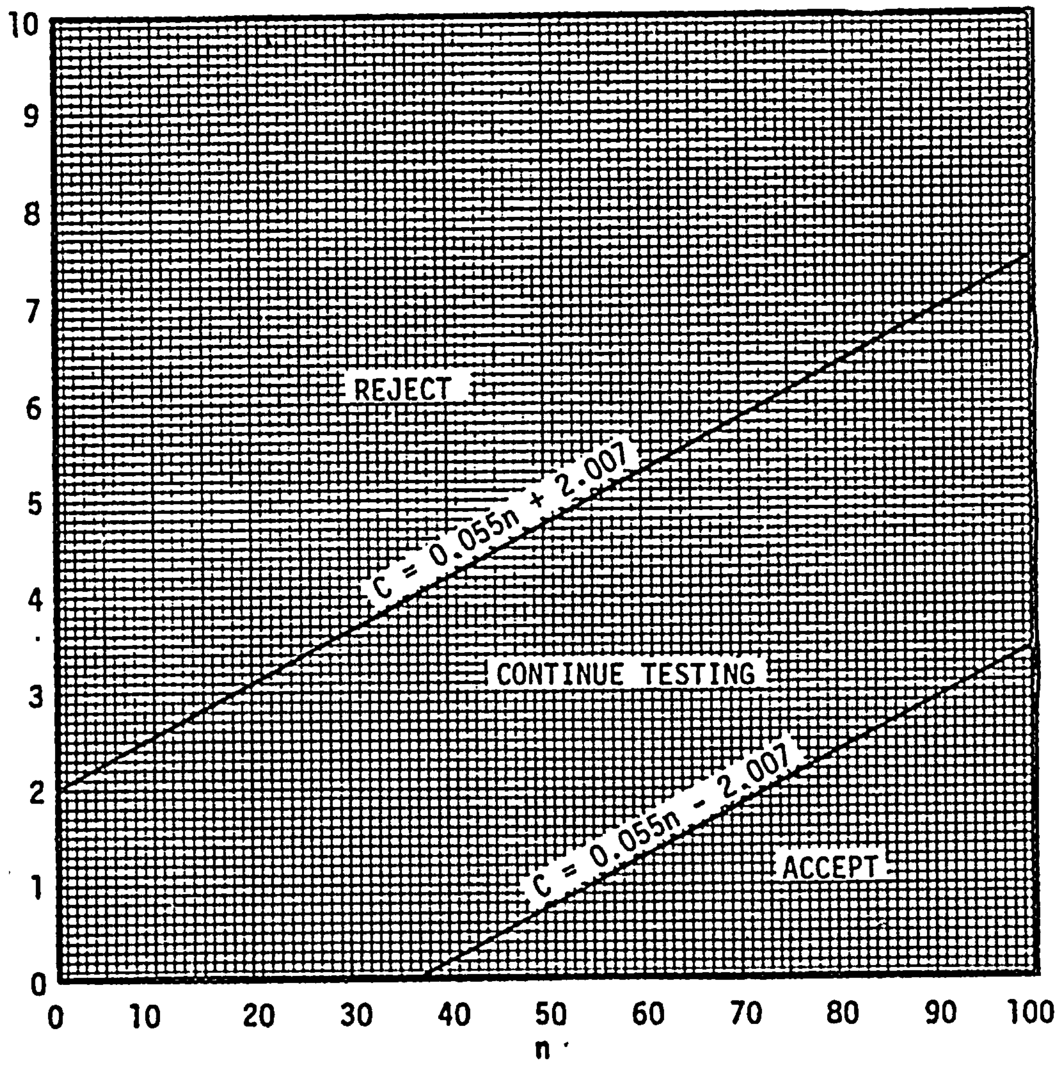

Figure 4.7.5-1 Sample Plan 2 For Snubber Functional Test

PLANT SYSTEMS

3/4.7.6 SEALED SOURCE CONTAMINATION

LIMITING CONDITIONS FOR OPERATION

3.7.6 Each sealed source containing radioactive material either in excess of 100 microcuries of beta- and/or gamma-emitting material or 5 microcuries of alpha-emitting material shall be free of 0.005 microcurie or more of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

    1. Decontaminate and repair the sealed source, or
    2. Dispose of the sealed source in accordance with Commission regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.6.1 Test Requirements

Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or
b. Other persons specifically authorized by the Commission or an Agreement State.

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

4.7.6.2 Test Frequencies

Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in Use

    At least once per 6 months for all sealed sources containing radioactive material:

    1. With a half-life greater than 30 days, excluding hydrogen-3, and
    2. In any form other than gas.

SURVEILLANCE REQUIREMENTS

4.7.6.2 (Continued)

b. Stored Sources Not in Use

Each sealed source and fission detector shall be tested before use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested before being placed into use.

c. Startup Sources and Fission Detectors

Each sealed startup source and fission detector shall be tested within 31 days before being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.6.3 Reports

A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of 0.005 microcurie or more of removable contamination.

PLANT SYSTEMS

3/4.7.7 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.7.7 The main turbine bypass system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 when THERMAL POWER is 25% or more of RATED THERMAL POWER.

ACTION:

With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.7.7 The main turbine bypass system shall be demonstrated OPERABLE at least once per 18 months by:

a. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve actuates to its correct position.

b. Demonstrating TURBINE BYPASS SYSTEM RESPONSE TIME meets the following requirements when measured from initial movement of the main turbine stop or control valve:

   1. 80% of the turbine bypass system capacity shall be established within 0.3 second, and

   2. Bypass valve opening shall start in less than or equal to 0.1 second.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 AC SOURCES

AC SOURCES - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 As a minimum, the following AC electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

   1. Separate day fuel tanks containing a minimum of 409 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II), and 282 gallons for EDG*2 (HPCS-Division III)

   2. A separate fuel storage system containing a minimum of 52,664 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II), and 36,173 gallons for EDG*2 (HPCS-Division III), and

   3. Two fuel oil transfer pumps.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one offsite circuit of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirements 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If either diesel generator EDG*1 or EDG*3 has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for each such diesel generator, separately, within 24 hours unless the diesel generators are already operating and loaded. Restore the offsite circuit 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 either diesel generator EDG*1 or EDG*3 inoperable, demonstrate the OPERABILITY of the above required AC offsite sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If the diesel generator became inoperable from any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators, separately, by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5

AC SOURCES - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 (Continued)

ACTION:

b. (Continued)

separately for each diesel generator within 24 hours.* Restore the inoperable diesel generator 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.

c. With one offsite circuit of the above required AC sources and diesel generator EDG*1 or EDG*3 of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If a diesel generator became inoperable from any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators, separately for each diesel generator, by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours for each diesel generator which has not been successfully tested in the past 24 hours unless the diesel generators are already operating and loaded.* Restore at least one of the inoperable AC 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 EDG*1 and EDG*3 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With diesel generator EDG*2 of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the offsite AC sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter. If the diesel generator becomes inoperable as a result of any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators, separately, by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours.* Restore diesel generator EDG*2 to OPERABLE status within 72 hours or declare the HPCS inoperable and take the ACTION required by Specifications 3.5.1 and 3.7.1.1.

* This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 (Continued)

ACTION:

e. With diesel generator EDG*1 or EDG*3 of the above required AC electrical power sources inoperable, in addition to taking ACTION 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.

f. With both of the above required offsite circuits inoperable, demonstrate the OPERABILITY of three diesel generators, separately, by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 separately for each diesel generator within 8 hours unless the diesel generators are already operating and loaded; restore at least one of the above required 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. A successful test(s) of diesel generator OPERABILITY per Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5, performed under this ACTION statement for the OPERABLE diesel generators, satisfies the diesel generator test requirements of ACTION statement a.

g. With diesel generators EDG*1 and EDG*3 of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once every 8 hours thereafter and Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for diesel generator EDG*2 within 8 hours.* Restore at least one of the inoperable diesel generators EDG*1 and EDG*3 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 EDG*1 and EDG*3 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABLE status. The provisions of Specification 3.0.2 are not applicable.

LIMITING CONDITIONS FOR OPERATION

3.8.1.1 (Continued)

ACTION:

h. With one offsite circuit of the above-required AC electrical power sources inoperable and diesel generator EDG*2 inoperable, apply the requirements of ACTIONS a and d specified above.

i. With either diesel generator EDG*1 or EDG*3 inoperable and diesel generator EDG*2 inoperable, apply the requirements of ACTIONS b, d, and e specified above.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once every 7 days by verifying correct breaker alignments and indicated power availability.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day fuel tank.

   2. Verifying the fuel level in the fuel storage tank.

   3. Verifying each fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.

   4. Verifying that on a start from ambient conditions:

      a) That diesel engines EDG*1 and EDG*3 accelerate to at least 600 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the start signal.

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* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.a.4 (Continued)

b) That diesel engine EDG*2 accelerates to at least 870 rpm and at least 3750 volts in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the start signal.

c) Each diesel generator shall be started for this test by using one of the following signals:

1) Manual.

2) Simulated loss of offsite power by itself.

3) Simulated loss of offsite power in conjunction with an ESF actuation test signal.

4) An ESF actuation test signal by itself.

5. Verifying that after the diesel generator is synchronized, it is loaded to greater than or equal to 4400 kW for diesel generators EDG*1 and EDG*3 and greater than or equal to 2600 kW for diesel generator EDG*2 in less than or equal to 90 seconds* and operates with these loads for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

7. Verifying the pressure in diesel generator air start receivers for EDG*1, EDG*2 and EDG*3 to be greater than or equal to 225 psig.

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* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

b. By removing accumulated water:

    1. From the day tank at least once per 31 days and after each occasion when the diesel is operated for more than 1 hour, and

    2. From the storage tank at least once per 31 days.

c. By sampling new fuel oil in accordance with ASTM D4057-81 before addition to the storage tanks and:

    1. By verifying in accordance with the tests specified in ASTM D975-81 before addition to the storage tanks that the sample has:

        a) An API gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees,

        b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification,

        c) A flash point equal to or greater than 125°F, and

        d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

    2. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81, except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. At least once per 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2 (Continued)

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 1125 kW for diesel generator EDG*1, greater than or equal to 750 kW for diesel generator EDG*3, and greater than or equal to 2433 kW for diesel generator EDG*2 while maintaining engine speed increase less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint or 15% of nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 4400 kW for diesel generators EDG*1 and EDG*3 and 2600 kW for diesel generator EDG*2 without tripping.** The generator voltage shall not exceed 4576 volts for EDG*1 and EDG*3, and 5824 volts for EDG*2 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 buses and load shedding from the emergency buses.

2) Verifying the diesel generator starts*** on the autostart signal, energizes the emergency buses with permanently connected loads within 13 seconds†, energizes the auto-connected (shutdown) loads through the load timers 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 buses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

* For any start of a diesel, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

** Momentary transients due to changing bus loads shall not invalidate the test.

*** All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

† From initiation of loss of offsite power.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e.4 (Continued)

b) For Division III:

1) Verifying deenergization of the emergency bus.

2) Verifying the diesel generator starts* on the autostart signal, energizes the emergency bus with the permanently connected loads within 13 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 ± 416 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power:

a) That diesel generators EDG*1 and EDG*3 start* on the autostart signal and operate on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 seconds after the autostart signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

b) That diesel generator EDG*2 starts* on the autostart signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the autostart signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Furthermore all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

** From initiation of loss of offsite power.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e (Continued)

6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) For Divisions I and II:

      1) Verifying deenergization of the emergency buses and loads shedding from the emergency buses.

      2) Verifying the diesel generator starts* on the autostart signal, energizes the emergency buses with permanently connected loads within 10 seconds, energizes the auto-connected (shutdown) loads through the load timers, 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 buses shall be maintained at 4160 ± 416 volts and 60 ± 1.2 Hz during this test.

   b) For Division III:

      1) Verifying deenergization of the emergency bus.

      2) Verifying the diesel generator starts* on the autostart signal, energizes the emergency bus with the permanently connected 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 ± 416 volts and 60 ± 1.2 Hz during this test.

7. 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 engine overspeed trip and generator differential trip.

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* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Furthermore, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures and may also include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e (Continued)

8. Verify the diesel generator operates for at least 24 hours.

   a) For Divisions I and II:

      During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 4840 kW*. During the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4400 kW*. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 3.0 Hz within 10 seconds and 4160 ± 416 volts and 60 ± 1.2 Hz within 13 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).**

   b) For Division III:

      During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2860 kW*. During the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 2600 kW*. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 15 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform surveillance requirement 4.8.1.1.2.e.4.b)2).**

9. Verifying that the autoconnected loads to each diesel generator do not exceed the 2000-hour rating of 4750 kW for diesel generators EDG*1 and EDG*3 and 2850 kW for diesel generator EDG*2.

10. Verifying the diesel generator's capability to:

   a) Manually 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.

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* Momentary transients due to changing bus loads shall not invalidate the test.

** If Surveillance Requirement 4.8.1.1.2.e.4.a)2) and/or 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 4400 kW or more for EDG*1 and EDG*3 and 2600 kW or more for EDG*2 for 1 hour or until operating temperature has stabilized before reperforming Surveillance Requirements 4.8.1.1.2.e.4.a)2) and/or b)2).

SURVEILLANCE REQUIREMENTS

4.8.1.1.2.e (Continued)

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that the automatic load timer relays are OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators EDG*1 and EDG*3.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) For Divisions I and II, turning gear engaged and emergency stop.

    b) For Division III, engine in the maintenance mode and diesel generator lockout.

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 diesel generators EDG*1 and EDG*3 accelerate to at least 600 rpm and EDG*2 accelerates to at least 870 rpm in less than or equal to 10 seconds.

g. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section II Article IWD-5000.

4.8.1.1.3 All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2, within 30 days. Reports of diesel generator failures shall include the information recommended in Position C.3.b of RG 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 Position C.3.b of RG 1.108, Revision 1, August 1977.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS*	NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	≤ 4	At least once per 31 days
≥ 2**	≥ 5	At least once per 7 days

* Criteria for determining number of failures and number of valid tests shall be in accordance with Position C.2.e of RG 1.108, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul, including appropriate postmaintenance operation and testing, is specifically approved by the manufacturer <u>and</u> if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 and four tests in accordance with the 184-day testing requirement of Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

** The associated test frequency shall be maintained until seven consecutive failure-free demands have been performed <u>and</u> the number of failures in the last 20 valid demands has been reduced to 1.

AC SOURCES

AC SOURCES - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.1.2 As a minimum, the following AC 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 EDG*1 or EDG*3, and diesel generator EDG*2 when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. Separate day fuel tanks containing a minimum of 409 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II) and 282 gallons for EDG*2 (HPCS-Division III).

   2. A separate fuel storage system containing a minimum of 52,664 gallons of fuel for EDG*1 (Division I) and EDG*3 (Division II) and 36,173 gallons of fuel for EDG*2 (HPCS-Division III).

   3. Two fuel oil transfer pumps.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

a. With less than the the above required AC electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are stored therein. In addition, in OPERATIONAL CONDITION 5, with the water level less than 22 feet 3 inches above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator EDG*2 of the above required AC electrical power sources inoperable, restore the inoperable diesel generator to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required AC electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2, and 4.8.1.1.3, except for the requirement of 4.8.1.1.2.a.5.

* When handling irradiated fuel in the secondary containment.

3/4.8.2 DC SOURCES

DC SOURCES - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.2.1 As a minimum, the following DC electrical power sources shall be OPERABLE:

a. Division I, consisting of:

1. 125-volt battery 2BYS*BAT 2A and
2. One 125-volt full-capacity charger*

b. Division II, consisting of:

1. 125-volt battery 2BYS*BAT 2B and
2. One 125-volt full-capacity charger*

c. Division III, consisting of:

1. 125-volt battery 2BYS*BAT 2C and
2. One 125-volt full-capacity 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 DC electrical power sources inoperable, restore the inoperable division DC electrical power source(s) 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 DC 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.

*Two 125-volt full capacity chargers are required when the UPS is powered by its backup DC power supply.

SURVEILLANCE REQUIREMENTS

4.8.2.1 (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 107 volts, or battery overcharge with battery terminal voltage above 142 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.
   3. The average electrolyte temperature of one out of five 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,

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 120% of the resistance readings taken during initial installation,* and

   4. The battery charger will supply:

      1. For Divisions I and II, at least 300 amperes at a minimum of 130 volts for at least 4 hours.

      2. For Division III, at least 40 amperes 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 2 hours for Divisions I and II, and 2 hours for Division III when the battery is subjected to a battery service test, or

   2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to 105 volts for Division I and II and 112.5 volts for Division III:

* In accordance with IEEE 450-1980.

SURVEILLANCE REQUIREMENTS

4.8.2.1.d.2 (Continued)

a) Division I. Greater than or equal to 818 amperes during the initial 60 seconds; greater than or equal to 445 amperes during the next 118 minutes; and greater than or equal to 701 amperes during the remainder of the 2-hour test.

b) Division II. Greater than or equal to 570 amperes during the initial 60 seconds; greater than or equal to 449 amperes during the next 118 minutes; and greater than or equal to 505 amperes during the remainder of the 2-hour test.

c) Division III. Greater than or equal to 54.6 amperes during the initial 60 seconds; greater than or equal to 15.4 amperes during the remainder of the 2-hour test.

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. During this once every 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(a)	CATEGORY B(b)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(c) 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(d)	>2.07 volts
Specific Gravity(e)	$\geq$1.200(f)	$\geq$1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells >1.205	Average of all connected cells $\geq$1.195(f)

TABLE 4.8.2.1-1 (Continued)

BATTERY SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(b) 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.

(c) Any Category B parameter not within its allowable value indicates an inoperable battery.

(d) May be corrected for average electrolyte temperature.

(e) Corrected for electrolyte temperature and level.

(f) Or battery charging current is less than 2 amperes when on float charge.

ELECTRICAL POWER SYSTEMS

DC SOURCES

DC SOURCES - SHUTDOWN

LIMITING CONDITIONS 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 DC electrical power sources shall be OPERABLE with:

a. Division I consisting of:

   1. 125-volt battery 2BYS*BAT 2A and
   2. One 125-volt full capacity charger.†

b. Division II consisting of:

   1. 125-volt battery 2BYS*BAT 2B and
   2. One 125-volt full capacity charger.†

c. Division III consisting of:

   1. 125-volt battery 2BYS*BAT 2C and
   2. One 125-volt full capacity 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 DC electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

b. With Division III battery and/or charger of the above required DC electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specifications 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.

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* When handling irradiated fuel in the secondary containment.

† Two 125 volt full capacity chargers are required when the UPS is powered by its backup DC power supply.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

DISTRIBUTION - OPERATING

LIMITING CONDITIONS FOR OPERATION

3.8.3.1 The following power distribution system divisions shall be energized with tie breakers open between Division I and Division II buses:

a. AC power distribution

1. Division I, consisting of:

    a) 4160-volt AC bus
    b) 600-volt AC load center/MCCs/distribution panels
    c) 240/120-volt AC and 120-volt AC distribution panels, energized from inverter 2VBA*UPS2A†

2. Division II, consisting of:

    a) 4160-volt AC bus
    b) 600-volt AC load center/MCCs/distribution panels
    c) 240/120-volt AC and 120-volt AC distribution panels, energized from inverter 2VBA*UPS2B†

3. Division III, consisting of:

    a) 4160-volt AC bus
    b) 600-volt AC MCCs/distribution panels
    c) 240/120-volt AC and 208/120-volt AC distribution panels

b. DC power distribution

1. Division I, consisting of 125-volt DC switchgear, MCC and associated distribution panels: 2BYS*PNL 201A; 2BYS*PNL 202A; 2BYS*PNL 204A

2. Division II, consisting of 125-volt DC switchgear, MCC and associated distribution panels: 2BYS*PNL 201B; 2BYS*PNL 202B; 2BYS*PNL 204B

3. Division III, consisting of 125-volt DC distribution panel 2CES*IPNL 414

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

† The UPS shall be energized from their normal AC supply or their backup DC supply.

LIMITING CONDITIONS FOR OPERATION

3.8.3.1 (Continued)

ACTION:

a. For AC power distribution:

1. With either Division I or Division II of the above required AC distribution system not energized, reenergize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division III of the above required AC distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

b. For DC power distribution:

1. With either Division I or Division II of the above required DC distribution system not energized, reenergize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division III of the above required DC distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.3.1.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct supply breaker alignment and by verifying no inoperability status indicator lights in the control room are lit.*

4.8.3.1.2 Each of the above required power distribution switchgear shall be determined energized at least once per 7 days by verifying the voltage on the panels.

* Which would indicate a loss of power to one or more of the required MCCs, load center, or panels.

DISTRIBUTION - SHUTDOWN

LIMITING CONDITIONS FOR OPERATION

3.8.3.2 As a minimum, the following power distribution system divisions shall be energized:

a. For AC 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 AC bus
      b) 600-volt AC load center/MCCs/distribution panels
      c) 240/120-volt AC and 120-volt AC distribution panels, energized from inverter 2VBA*UPS2A or alternate supply

   2. Division II consisting of:

      a) 4160-volt AC bus
      b) 600-volt AC load center/MCCs/distribution panels
      c) 240/120-volt AC and 120-volt AC distribution panels, energized from inverter 2VBA*UPS2B or alternate supply

   3. Division III consisting of:

      a) 4160-volt AC bus
      b) 600-volt AC MCCs/distribution panels
      c) 240/120-volt AC and 208/120-volt AC distribution panels

b. For DC 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 DC switchgear, MCC, and distribution panels

   2. Division II consisting of 125-volt DC switchgear, MCC, and distribution panels

   3. Division III consisting of 125-volt DC distribution panels

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

* When handling irradiated fuel in the reactor building.

LIMITING CONDITIONS FOR OPERATION

3.8.3.2 (Continued)

ACTION:

a. For AC power distribution:

   1. With less than Division I and Division II of the above required AC distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the reactor building, and operations with a potential for draining the reactor vessel.

   2. With Division III of the above required AC distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

b. For DC power distribution:

   1. With less than Division I and Division II of the above required DC distribution system energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the reactor building, and operations with a potential for draining the reactor vessel.

   2. With Division III of the above required DC distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2.1 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct supply breaker alignment and by verifying no inoperability status indicator lights in the control room are lit.*

4.8.3.2.2 Each of the above required power distribution switchgear shall be determined energized at least once per 7 days by verifying the voltage on the panels.

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* Which would indicate loss of power to one or more of the required MCCs, load centers, or panels.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

AC CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITIONS FOR OPERATION

3.8.4.1 The AC circuits inside primary containment shown in Table 3.8.4.1-1 shall be deenergized:*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required AC circuits shall be determined to be deenergized at least once every 24 hours** by verifying that the associated circuit breakers are in the tripped condition.

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* Required before power ascension and following final drywell inspection.

** Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

TABLE 3.8.4.1-1

PRIMARY CONTAINMENT AC CIRCUITS DEENERGIZED

CIRCUIT NO.	POWER SOURCE	CIRCUIT BKR NO.	EQUIPMENT POWERED
N12-11	2LAR-PNLN12	11	Normal Lighting Ckts - El 261'
N12-12	2LAR-PNLN12	12	Normal Lighting Ckts - El 261'
N12-13	2LAR-PNLN12	13	Normal Lighting Ckts - El 261'
N12-14	2LAR-PNLN12	14	Normal Lighting Ckts - El 289'
N12-15	2LAR-PNLN12	15	Normal Lighting Ckts - El 289'
N12-16	2LAR-PNLN12	16	Normal Lighting Ckts - El 289'
N12-17	2LAR-PNLN12	17	Normal Lighting Ckts - El 240'
N12-18	2LAR-PNLN12	18	Normal Lighting Ckts - El 240'
N12-19	2LAR-PNLN12	19	Normal Lighting Ckts - El 240'
N12-1	2LAR-PNLN12	1	Normal Receptacle Ckts - El 240'
N12-2	2LAR-PNLN12	2	Normal Receptacle Ckts - El 261'
N12-3	2LAR-PNLN12	3	Normal Receptacle Ckts - El 289'
N05-14	2LAR-PNLN05	14	SW for Normal Ltg Contractor Coil Ckt
U02-14	2LAR-PNLU02	14	Essential Lighting Ckts - El 240'
U02-15	2LAR-PNLU02	15	Essential Lighting Ckts - El 261'
U02-16	2LAR-PNLU02	16	Essential Lighting Ckts - El 289'
U02-12	2LAR-PNLU02	12	SW for Essential Ltg Contractor Coil Ckt
N03-7	2LAR-PNLN03	7	Normal Receptacle Ckts - Supp Pool
NA	2WPS-PNL200	20	2WPS-RCPT51A & B - Welding Recept El 261'
NA	2WPS-PNL200	22	2WPS-RCPT51A & B - Welding Recept El 261'
NA	2WPS-PNL200	24	2WPS-RCPT51A & B - Welding Recept El 261'
NA	2WPS-PNL200	26	2WPS-RCPT52A & B - Welding Recept El 261'
NA	2WPS-PNL200	28	2WPS-RCPT52A & B - Welding Recept El 261'
NA	2WPS-PNL200	30	2WPS-RCPT52A & B - Welding Recept El 261'

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT AC CIRCUITS DEENERGIZED

CIRCUIT NO.	POWER SOURCE	SECT	EQUIPMENT POWERED
2DERA03	2NHS-MCC012	7B	2DER*MOV128 - Reactor Drain Isol Valve
NA	2NHS-MCC005	7B	2MHR-CRN3 - Recirc Mtr Hndlg Crane - AMHR PNL101
NA	2NHS-MCC005	7C	2MHR-CRN4 - Recirc Mtr Hndlg Crane - 2MHR PNL102
NA	2NHS-MCC005	7D	2MHR-CRN65 - Monorail 2 Ton for 2MSS*PSV
NA	2NHS-MCC005	7E	2MHR-CRN66 - Monorail 2 Ton for RDS Cart
NA	2NHS-MCC005	7F	2MHR-CRN67 - Monorail 2 Ton for 2MSS*HVY Valves

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT AC CIRCUITS DEENERGIZED

CIRCUIT NO.	POWER SOURCE	CIRCUIT BKR NO.	EQUIPMENT POWERED
U01-15	2LAR-PNLU01	15	Comm-Party Paging Suppression Pool
U03-12	2LAR-PNLU03	12	Comm-Party Paging Above Suppression Pool

CIRCUIT NO.	MAINTENANCE/CALIBR SELECTOR SWITCH PANEL	SWITCH NO.	EQUIPMENT POWERED
NA	RSC-88	124	Maintenance/Calibration Jack - JK124
NA	RSC-88	128	Maintenance/Calibration Jack - JK128
NA	RSC-88	134	Maintenance/Calibration Jack - JK134
NA	RSC-88	137	Maintenance/Calibration Jack - JK137

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITIONS FOR OPERATION

3.8.4.2 All primary containment penetration conductor overcurrent protective devices* 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* inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

    1. For 13.8-kV circuit breakers, deenergize the 13.8-kV circuits by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker(s) to be tripped at least once every 7 days thereafter.

    2. For 600 volt MCC circuit breakers, remove the inoperable circuit breaker(s) from service by opening the breaker within 72 hours and verify the inoperable breaker(s) to be in the open position at least once every 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 13.8-kV circuits which have their redundant circuit breakers tripped or to 600-volt circuits which have the inoperable circuit breaker disconnected.

SURVEILLANCE REQUIREMENTS

4.8.4.2 Each of the primary containment penetration conductor overcurrent protective devices* shall be demonstrated OPERABLE:

a. At least once per 18 months:

    1. By verifying that the medium voltage 13.8-kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level and performing:

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* Excluded from this specification are those penetration assemblies that are capable of withstanding the maximum current available because of an electrical fault inside containment.

SURVEILLANCE REQUIREMENTS

4.8.4.2.a.1 (Continued)

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 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.

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 in excess 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 before 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.

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.

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

EMERGENCY LIGHTING SYSTEM - OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 The emergency lighting system overcurrent protection devices shown in Table 3.8.4.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With one or more of the overcurrent protective devices shown in Table 3.8.4.3-1 inoperable, within 72 hours remove the inoperable circuit breaker(s) from service by opening the breaker. Return the breaker(s) to OPERABLE status within 7 days, otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The overcurrent protective devices shall be demonstrated OPERABLE at least once per 18 months by selecting and testing one-half of each type of circuit breaker on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the time delay element. The measured response time shall be compared with the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of the nominal instantaneous pickup setting and verifying that circuit breaker trips instantaneously with no intentional time delay.

TABLE 3.8.4.3-1

OVERCURRENT PROTECTIVE DEVICES FOR

NON-CLASS 1E LIGHTING FIXTURES ON CLASS 1E EMERGENCY SYSTEM

PRIMARY CIRCUIT BREAKER			BACKUP CIRCUIT BREAKER			
MFR/TYPE	CURRENT RATING	EQUIPMENT POWERED LOCATION - 120/208-V	MFR/TYPE	CURRENT RATING	CKT NO.	POWER SUPPLY LOCATION - 600V
Gould/EH	100A	2LAC*PNLE01 Div I Swgr Diesel Gen Rm & Remote Shutdown Rm	Gould/HE	45A	1	2LAC*PNL100A
Gould/EH	100A	2LAC*PNLE04 Relay Room	Gould/HE	45A	2	2LAC*PNL100A
Gould/EH	100A	2LAC*PNLE06 Control Room	Gould/HE	45A	8	2LAC*PNL100A
Gould/EH	100A	2LAC*PNLE02 Div II Swgr Diesel Gen Rm & Remote Shutdown Rm	Gould/HE	45A	1	2LAC*PNL300B
Gould/EH	100A	2LAC*PNLE05 Relay Room	Gould/HE	45A	2	2LAC*PNL300B
Gould/EH	100A	2LAC*PNLE07 Control Room	Gould/HE	45A	8	2LAC*PNL300B
Gould/EH	100A	2LAC*PNLE03 Div III Swgr Diesel Gen Rm	Gould/HE	45A	Compt 10B	2EHS*MCC201

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING (RPS LOGIC)

LIMITING CONDITIONS FOR OPERATION

3.8.4.4 Two RPS UPS electrical protection assemblies for each inservice UPS set or alternate source shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electrical protection assembly for an inservice RPS UPS inoperable, restore the inoperable electrical protection assembly to OPERABLE status within 72 hours or remove the associated RPS UPS from service.

b. With both RPS electrical protection assemblies for an inservice RPS UPS inoperable, restore at least one electrical protection assembly to OPERABLE status within 30 minutes or remove the associated RPS UPS from service.

SURVEILLANCE REQUIREMENTS

4.8.4.4 The above specified RPS electrical protection assemblies instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST and;

b. At least once per 18 months by demonstrating the OPERABILITY of overvoltage, undervoltage and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

   1. Overvoltage Bus A: $\leq$ 132 volts AC
      Bus B: $\leq$ 132 volts AC

   2. Undervoltage Bus A: $\geq$ 117.1 volts AC
      Bus B: $\geq$ 115.75 volts AC

   3. Underfrequency $\geq$57 Hz

ELECTRICAL POWER SYSTEMS

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

REACTOR PROTECTION SYSTEM ELECTRIC POWER MONITORING (SCRAM SOLENOIDS)

LIMITING CONDITIONS FOR OPERATION

3.8.4.5 Two RPS electrical protection assemblies (EPAs) for each inservice RPS MG set or alternate source shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one RPS electrical protection assembly for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable EPA to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

b. With both RPS electrical protection assemblies for an inservice RPS MG set or alternate power supply inoperable, restore at least one EPA to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.5 The above specified RPS electrical protection assemblies shall be determined OPERABLE:

a. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST and;

b. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, undervoltage and underfrequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Overvoltage Bus A: $\leq$ 128.8 volts AC
   Bus B: $\leq$ 130.0 volts AC

2. Undervoltage Bus A: $\geq$ 114.5 volts AC
   Bus B: $\geq$ 115.1 volts AC

3. Underfrequency $\geq$57 Hz

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITIONS FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform hoists fuel-loaded.
4. Fuel grapple position.
5. Service platform hoist fuel-loaded.

APPLICABILITY: OPERATIONAL CONDITION 5* #

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

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* 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.

REFUELING OPERATIONS

REACTOR MODE SWITCH

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 before:

   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 before 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 before resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

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* 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 CONDITIONS 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. Audible indication in the control room,

c. 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

d. Unless adequate shutdown margin has been demonstrated per Specification 3.1.1 and the "one rod out" interlock is OPERABLE per Specification 3.9.1, the shorting links shall be removed from the RPS circuitry prior to and any time one 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. Performing a CHANNEL CHECK,

2. Verifying the detectors are inserted to the normal operating level, and

3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

---

* The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

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

SURVEILLANCE REQUIREMENTS

4.9.2 (Continued)

b. Performing a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours before the start of CORE ALTERATIONS, and
   2. At least once per 7 days.

c. Verifying that the channel count rate is at least 3 cps*

   1. Before control rod withdrawal,

   2. Before and at least once per 12 hours during CORE ALTERATIONS, and

   3. At least once per 24 hours.

d. Verifying, within 8 hours before and at least once per 12 hours during the time any control rod is withdrawn that the shorting links have been removed from the RPS circuitry, unless adequate shutdown margin has been demonstrated per Specification 3.1.1 and the "one rod out" interlock is OPERABLE per Specification 3.9.1.

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* For initial loading and startup the count rate may be less than 3 cps if the following conditions are met: (1) the signal-to-noise ratio is greater than or equal to 20 and (2) the signal is greater than 0.7 cps.

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITIONS FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified:

a. Within 2 hours before:

   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, or with one control rod withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

** See Special Test Exception 3.10.3.

3/4.9.4 DECAY TIME

LIMITING CONDITIONS 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 before movement of irradiated fuel in the reactor pressure vessel.

REFERENCE OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITIONS FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling floor personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.

ACTION:

When direct communication between the control room and refueling floor personnel cannot be maintained, immediately suspend CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated within 1 hour before the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 REFUELING PLATFORM

LIMITING CONDITIONS FOR OPERATION

3.9.6 The refueling platform shall be OPERABLE and used for handling fuel assemblies or control rods within the reactor pressure vessel.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel.

ACTION:

With the requirements for refueling platform OPERABILITY not satisfied, suspend use of any inoperable refueling platform equipment from operations involving the handling of control rods and fuel assemblies within the reactor pressure vessel after placing the load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.6 Each refueling platform crane or hoist used for handling of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 7 days before the start of such operations with that crane or hoist by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1200 ± 50 pounds.

b. Demonstrating operation of the overload cutoff on the frame mounted and monorail mounted auxiliary hoists when the load exceeds 1000 ± 50 pounds.

c. Demonstrating operation of the main and auxiliary hoist uptravel stops when the grapple is lower than or equal to 7 feet 9 3/4 inches below the platform tracks.

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 POOL

LIMITING CONDITIONS FOR OPERATION

3.9.7 Loads in excess of 1000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks that prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days before and at least once per 7 days during crane operation with loads in excess of 1000 pounds.

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 3 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours before the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITIONS FOR OPERATION

3.9.9 At least 22 feet 3 inches of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

REFUELING OPERATIONS

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITIONS 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 5 x 5 array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

REFUELING OPERATIONS

CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours before 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 every 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 5 x 5 array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITIONS 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 (SRMs) 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 have been suspended.

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.

REFUELING OPERATIONS

CONTROL ROD REMOVAL

MULTIPLE CONTROL ROD REMOVAL

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours before 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 have been suspended.

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 CONDITIONS FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation* with at least:

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 3 inches above the top of the reactor pressure vessel flange.

ACTION:

a. With no RHR shutdown cooling mode loop OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

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

* The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

LOW WATER LEVEL

LIMITING CONDITIONS FOR OPERATION

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

a. One OPERABLE RHR pump, and
b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet 3 inches above the top of the reactor pressure vessel flange.

ACTION:

a. With less than the above required shutdown cooling mode loops of the RHR system OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternative method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternative 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.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3, and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low-power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during low-power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod sequence control system (RSCS) per Specification 3.1.4.2 may be suspended by means of bypass switches for the following tests provided that the rod worth minimizer is OPERABLE per Specification 3.1.4.1:

a. Shutdown margin demonstrations, Specification 4.1.1.
b. Control rod scram, Specification 4.1.3.2.
c. Control rod friction measurements.
d. Startup Test Program with the THERMAL POWER less than 20% of RATED THERMAL POWER.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the requirements of the above specification not satisfied, verify that the RSCS is OPERABLE per Specification 3.1.4.2.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed on control rod groups by the RSCS are bypassed, verify:

a. Within 8 hours before bypassing any sequence constraint and at least once per 12 hours while any sequence constraint is bypassed:

    1. That the rod worth minimizer is OPERABLE per Specification 3.1.4.1,
    2. That movement of control rods from 75% ROD DENSITY to the RSCS low-power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests.

b. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

SPECIAL TEST EXCEPTION

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITIONS FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3, and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The continuous rod 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,

b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITIONS 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.

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITIONS FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.2 may be suspended during the performance of the Startup Test Program until either the required 100% of RATED THERMAL POWER trip tests have been completed or the reactor has operated for 120 effective full-power days.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The effective full-power days of operation shall be verified to be less than 120, by calculation, at least once per 7 days during the Startup Test Program.

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITIONS FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during training startups.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

LIMITING CONDITIONS FOR OPERATION

3.10.7 During initial core loading within the Startup Test Program the provisions of Specification 3/4.9.2 may be suspended provided that at least two source range monitor (SRM) channels with detectors inserted to the normal operating level are OPERABLE with:

a. One of the required SRM channels continuously indicating* 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,**

c. The RPS "shorting links" shall be removed prior to and during fuel loading,

d. The reactor mode switch is OPERABLE and locked in the Refuel position.

APPLICABILITY: OPERATIONAL CONDITION 5

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving initial core loading.

SURVEILLANCE REQUIREMENTS

4.10.7.1 Within one hour prior to and at least once per 12 hours during the initial core loading verify that:

a. The above required SRM channels are OPERABLE by:

1. Performance of a CHANNEL CHECK***

2. Confirming that the above required SRM detectors are at the normal operating level and located in the quadrants required by Specification 3.10.7.

*Up to 16 fuel bundles may be loaded without a visual indication of count rate.

**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.

***May be performed by use of movable neutron source.

SURVEILLANCE REQUIREMENTS (Continued)

4.10.7.1 (Continued)

b. The RPS "shorting links" are removed.

c. The reactor mode switch is locked in the REFUEL position.

4.10.7.2 Perform a CHANNEL FUNCTIONAL TEST for the above required SRM channels within 24 hours prior to the start and at least once per 7 days during initial core loading.

4.10.7.3 For at least one SRM channel, verify that the count rate is at least 0.7 cps*:

a. Immediately following the loading of the first 16 fuel bundles.

b. At least once per 12 hours thereafter during initial core loading.

*Provided signal-to-noise ratio is $\geq$ 20. Otherwise, 3 cps.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITIONS FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited to the concentrations specified in 10 CFR 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} microcurie/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.

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.

TABLE 4.11.1-1

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)
1. Batch Waste Release Tanks(b)	P Each Batch	P Each Batch	Principal Gamma Emitters(c)	$5x10^{-7}$
a. 2LWS-TK4A b. 2LWS-TK4B c. 2LWS-TK5A d. 2LWS-TK5B			I-131	$1x10^{-6}$
	P One Batch/M	One Batch/M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P Each Batch	M Composite(d)	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite(d)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
2. Continuous Releases	Grab Sample M(e)	Grab Sample M(e)	Principal Gamma Emitters(c)	$5x10^{-7}$
a. Service Water Effluent A			I-131	$1x10^{-6}$
b. Service Water Effluent B			Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
c. Cooling Tower Blowdown			H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Grab Sample Q(e)	Grab Sample Q(e)	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
d. Auxiliary Boiler Pump Seal and Sample Cooling Discharge (Service Water)	Grab Sample M(f)	Gram Sample M(f)	Principal Gamma Emitters(c)	$5x10^{-7}$
	Grab Sample Q(f)	Grab Sample Q(f)	H-3	$1x10^{-5}$

TABLE 4.11.1-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATIONS

(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:

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the before-the-fact lower limit of detection (microcurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microcurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (seconds).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system and not as an after-the-fact limit for a particular measurement.

(b) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

TABLE 4.11.1-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATIONS

(c) The principal gamma emmiters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137 and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6}. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8 in the format outlined in RG 1.21, Appendix B, Revision 1, June 1974.

(d) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(e) If the alarm setpoint of the effluent monitor, as determined by the method presented in the ODCM, is exceeded, the frequency of sampling shall be increased to daily until the condition no longer exists. Frequency of analysis shall be increased to daily for principal gamma emitters and an incident composite for H-3, gross alpha, Sr-89, Sr-90, and Fe-55.

(f) If the alarm setpoint of Service Water Effluent Monitor A and/or B, as determined by the method presented in the ODCM, is exceeded, the frequency of sampling shall be increased to daily until the condition no longer exists. Frequency of analysis shall be increased to daily for principal gamma emitters and an incident composite for H-3, gross alpha, Sr-89, Sr-90, and Fe-55.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

DOSE

LIMITING CONDITIONS FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the whole body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the whole body and to less than or equal to 10 mrem 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.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE, and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from the unit, to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.06 mrem to the whole 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 and any portion of the liquid radwaste treatment system not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when liquid radwaste treatment systems are not being fully utilized.

4.11.1.3.2 The installed liquid radwaste treatment system shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITIONS FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each 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 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, pursuant to Specification 6.9.1.8.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each 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.

* Tanks included in this specification are those 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, such as temporary tanks.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITIONS FOR OPERATION

3.11.2.1 The dose rate from radioactive materials released in gaseous effluents from the site to areas at or 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 mrem/yr to the whole body and less than or equal to 3000 mrem/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 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate from noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11.2-1.

TABLE 4.11.2-1

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)
1. Containment(b)	Each PURGE	P	Principal Gamma Emitters(c)	1x10-4
		Each PURGE	H-3 (oxide), Principal Gamma Emitters(c)	1×10^{-6}, 1×10^{-4}
2. Main Stack Radwaste/Reactor Building Vent	M(d)	M(d)	Principal Gamma Emitters(c)	1×10^{-4}
	Grab Sample M(e)	M(e)	H-3 (oxide)	1×10^{-6}
	Continuous(f)	W(g) Charcoal Sample	I-131	1×10^{-12}
	Continuous(f)	W(g) Particulate Sample	Principal Gamma Emitters(c) Gross Alpha	1×10^{-11} 1×10^{-11}
	Continuous(f)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

TABLE 4.11.2-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATIONS

(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 = the before-the-fact lower limit of detection (microcuries per unit mass or volume)

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute)

E = the counting efficiency (counts per disintegration)

V = the sample size (units of mass or volume)

2.22×10^6 = the number of disintegrations per minute per micro curie

Y = the fractional radiochemical yield, when applicable

λ = the radioactive decay constant for the particular radionuclide (sec^{-1})

Δt = the elapsed time between the midpoint of sample collection and the time of counting (seconds)

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system and not as an after-the-fact limit for a particular measurement.

TABLE 4.11.2-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATIONS

(b) Sample and analysis before PURGE is used to determine permissible PURGE rates. Sample and analysis during actual PURGE is used for offsite dose calculations.

(c) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8 in the format outlined in RG 1.21, Appendix B, Revision 1, June 1974.

(d) If the main stack or reactor/radwaste building isotopic monitor is not OPERABLE, sampling and analysis shall also be performed following shutdown, startup, or when there is an alert alarm on the offgas pretreatment monitor.

(e) Tritium grab samples shall be taken weekly from the reactor/radwaste ventilation system when fuel is offloaded until stable tritium release levels can be demonstrated.

(f) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1.b and 3.11.2.3.

(g) When the release rate of the main stack or reactor/radwaste building vent exceeds its alert alarm setpoint, the iodine and particulate device shall be removed and analyzed to determine the changes in iodine and particulate release rates. The analysis shall be done daily until the release no longer exceeds the alarm setpoint. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITIONS FOR OPERATION

3.11.2.2 The air dose from noble gases released in gaseous effluents, from each unit, to areas at or 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 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad 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.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITIONS FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radioactive material in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at or beyond the SITE BOUNDARY (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radioactive material 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(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

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radioactive material 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.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT 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 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

4.11.2.4 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.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

VENTILATION EXHAUST TREATMENT SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.11.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and appropriate portions of this system shall be used to reduce releases of radioactivity when the projected doses in 31 days from iodine and particulate releases, from each unit, to areas at or beyond the SITE BOUNDARY (see Figure 5.1.3-1) would exceed 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous 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. 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 from iodine and particulate releases from each unit to areas at or beyond the SITE BOUNDARY 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 being fully utilized.

4.11.2.5.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 or 3.11.2.3.

LIMITING CONDITIONS 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: At all times.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

b. With continuous monitors inoperable, utilize grab sampling procedures for a period not to exceed 30 days.

c. 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 sytem is in operation with the hydrogen monitors required OPERABLE by Table 3.3.7.10-1 of Specification 3.3.7.10.

*The offgas system is not required to be OPERABLE prior to the initial opening of the main steam isolation valves in OPERATIONAL CONDITION 2.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

MAIN CONDENSER - OFFGAS

LIMITING CONDITIONS FOR OPERATION

3.11.2.7 The radioactivity rate of noble gases measured downstream of the recombiner shall be limited to less than or equal to 350,000 microcuries/second.

APPLICABILITY: During offgas system operation.

ACTION:

With the radioactive rate of noble gases downstream of the recombiner exceeding 350,000 microcuries/second, restore the radioactivity 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 radioactive rate of noble gases downstream of the recombiner shall be continuously monitored in accordance with Specification 3.3.7.10.

4.11.2.7.2 The radioactivity rate of noble gases downstream of the recombiner shall be determined to be within the limits of Specifications 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken before holdup and discharge downstream of the recombiner:

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the offgas 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.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

VENTING OR PURGING

LIMITING CONDITIONS FOR OPERATION

3.11.2.8 VENTING or PURGING of the drywell and/or suppression chamber shall be through the standby gas treatment system.*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING and PURGING of the drywell and/or suppression chamber.

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

SURVEILLANCE REQUIREMENTS

4.11.2.8.1 The drywell and/or suppression chamber shall be determined to be aligned for VENTING or PURGING through the standby gas treatment system within 4 hours before start of and at least once per 12 hours during VENTING or PURGING.

* See Specification 3.6.5.3.

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITIONS FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three 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; and

SURVEILLANCE REQUIREMENTS

4.11.3 (Continued)

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITIONS 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 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

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 shall be made including direct radiation contributions from the units (including outside storage tanks, etc.) to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405(c), 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 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 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 units (including outside storage tanks, etc.) shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a of Specification 3.11.4.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITIONS FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 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 from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 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 required by Specification 6.9.1.7.

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* The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITIONS FOR OPERATION

3.12.1 (Continued)

ACTION:

c. With milk or fresh leafy vegetation samples unavailable from one or more of the sample locations required by Table 3.12.1-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.8, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

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 SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(b)	32 routine monitoring stations either with 2 or more dosimeters or with 1 instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY An outer ring of stations, one in each land base meteorological sector in the 4- to 5-mile* range from the site The balance of the stations should be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.(c)	Once per 3 months	Gamma dose once per 3 months

* At this distance, 8 windrose sectors (W, WNW, NW, NNW, N, NNE, NE, and ENE) are over Lake Ontario.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne Radioiodine and Particulates	Samples from five locations: 3 samples from offsite locations close to the site boundary (within one mile) in different sectors of the highest calculated annual site average ground-level D/Q (based on all site licensed reactors) 1 sample from the vicinity of an established year-round community having the highest calculated annual site average ground-level D/Q (based on all site licensed reactors) 1 sample from a control location, at least 10 miles distant and in a least prevalent wind direction(c)	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	Radioiodine Canister I-131 analysis weekly Particulate Sampler Gross beta radioactivity analysis following filter change(d) and gamma isotopic analysis(e) of composite (by location) at least quarterly
3. Waterborne			
a. Surface(f)	One sample upstream(c); one sample from the site's downstream cooling water intake	Composite sample over 1-month period(g)	Gamma isotopic analysis(e) once/month; composite for tritium analysis once/3 months

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterbone (Continued)			
b. Ground	Samples from one or two sources, only if likely to be affected(h)	Quarterly grab sample	Gamma isotopic(e) and tritium analysis quarterly
c. Drinking	1 sample of each of one to three of the nearest water supplies that could be affected by its discharge(i)	Composite sample over a 2-week period(g) when I-131 analysis is performed; monthly composite otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.(j) Composite for gross beta and gamma isotopic analyses(e) monthly. Composite for tritium analysis quarterly
d. Sediment from Shoreline	1 sample from a downstream area with existing or potential recreational value	Twice per year	Gamma isotopic analysis(e)

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion			
a. Milk	Samples from MILK SAMPLING LOCATIONS in three locations within 3.5 miles distance having the highest calculated site average D/Q (based on all licensed site reactors). If there are none, then 1 sample from MILK SAMPLING LOCATIONS in each of three areas 3.5-5.0 miles distant having the highest calculated site average D/Q (based on all licensed site reactors). One sample from a MILK SAMPLING LOCATION at a control location 9-20 miles distant and in a least prevelant wind direction(c)	Twice per month, April-December (samples will be collected January-March if I-131 is detected in November and December of the preceding year)	Gamma isotopic(e) and I-131 analysis twice/month when animals are on pasture (April-December); once per month at other times (January-March if required)
b. Fish	One sample each of two commercially or recreationally important species in the vicinity of a plant discharge area(k) One sample of the same species in areas not influenced by station discharge(c)	Twice per year	Gamma isotopic analysis(e) on edible portions twice per year

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS(a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged(l)	At time of harvest(m)	Gamma isotopic(e) analysis of edible portions (isotopic to include I-131)
	Samples of three different kinds of broad leaf vegetation (such as vegetables) grown nearest to each of two different offsite locations of highest calculated site average D/Q (based on all licensed site reactors)	Once per year during the harvest season	Gamma isotopic(e) analysis of edible portions (isotopic to include I-131)
	One sample of each of the similar broad leaf vegetation grown at least 9.3 miles distant in a least prevalent wind direction	Once per year during the harvest season	Gamma isotopic(e) analysis of edible portions (isotopic to include I-131)

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATIONS

(a) Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable because of such circumstances as hazardous conditions, seasonal unavailability,* or malfunction of automatic sampling equipment. If specimens are unobtainable because sampling equipment malfunctions, effort shall be made to complete corrective action before the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. It is recognized that, at times, it may not be possible or practical 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 may be made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. Pursuant to Specification 6.9.1.8, submit in the next Semiannual Radioactive Effluent Release Report a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of new location(s) for obtaining samples.

(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.

(c) The purpose of these samples 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, which 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 10 times the previous yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(e) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

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*Seasonal unavailability is meant to include theft and uncooperative residents.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATIONS

(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.

(g) In this program, representative 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 (refer to the ODCM for definition of representative composite 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 (see ODCM for discussion).

(i) Drinking water samples shall be taken only when drinking water is a dose pathway (see ODCM for discussion).

(j) Analysis for I-131 may be accomplished by Ge-Li analysis provided that the lower limit of detection (LLD) for I-131 in water samples found on Table 4.12.1-1 can be met. Doses shall be calculated for the maximum organ and age group; using the methodology in the ODCM.

(k) In the event two commercially or recreationally important species are not available, after three attempts of collection, then two samples of one species or other species not necessarily commercially or recreationally important may be utilized.

(l) This specification applies only to major irrigation projects within 9 miles of the site in the general "downcurrent" direction (see ODCM for discussion).

(m) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be taken monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12.1-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

RADIONUCLIDE ANALYSIS	WATER (pCi/1)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/1)	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-95, 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 a 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000 pCi/liter may be used.

** If no drinking water pathway exists, a value of 20 pCi/liter may be used.

TABLE 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(a)(b)

LOWER LIMIT OF DETECTION(c)

RADIONUCLIDE ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-95, Nb-95	15					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba/La-140	15			15		

*If no drinking water pathway exists, a value of 3000 pCi/liter may be used.

**If no drinking water pathway exists, a value of 15 pCi/liter may be used.

TABLE 4.12.1-1 (Continued)

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE

ANALYSIS - LOWER LIMIT OF DETECTION

TABLE NOTATIONS

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

(b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ANSI N-545, Section 4.3 1975. Allowable exceptions to ANSI N-545, Section 4.3 are contained in the Nine Mile Point Unit 2 ODCM.

(c) The lower limit of detection (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 = the before-the-fact lower limit of detection (picocuries per unit mass or volume)

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute)

E = the counting efficiency (counts per disintegration)

V = the sample size (units of mass or volume)

2.22 = the number of disintegrations per minute per picocurie

Y = the fractional radiochemical yield, when applicable

λ = the radioactive decay constant for the particular radionuclide (sec^{-1})

Δt = the elapsed time between environmental collection, or end of the sample collection period, and time of counting (seconds)

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12.1-1 (Continued)

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE

ANALYSIS - LOWER LIMIT OF DETECTION

TABLE NOTATIONS

It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system and not as an after-the-fact limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITIONS FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 5 miles the location in each of the 16 meteorological sectors of the nearest milk animal and the nearest residence, and the nearest garden* of greater than 500 square feet producing broad leaf vegetation. For elevated releases as defined in RG 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 3 miles the locations in each of the 16 meterological sectors of all milk animals and all gardens* greater than 500 square feet producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose, dose commitment, or D/Q value greater than the values currently being calculated in Specification 4.11.2.3, pursuant to Specification 6.9.1.8, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a land use census identifying a location(s) that yields a calculated dose, dose commitment, or D/Q value (via the same exposure pathway) significantly greater (50%) than at a location from which samples are currently being obtained in accordance with Specification 3.12.1-1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose, dose commitment(s) or D/Q value, via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.8 submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

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

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* Broad leaf vegetation sampling of at least three different kinds of vegetation, such as garden vegetables, may be performed at offsite locations in each of two different locations with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12.1-1, Part 4.c, shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

LAND USE CENSUS

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once every 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 6.9.1.7.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITIONS FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12.1-1. Participation in this program shall include media for which environmental samples are routinely collected and for which intercomparison samples are available.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

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 3.7.3 requires two control room outdoor air special filter train subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, measures must be initiated within 1 hour to place the unit in at least STARTUP within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours. As a further example, Specification 3.6.6.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, measures must be initiated within 1 hour 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.

APPLICABILITY

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL CONDITIONS or other 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 provides allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance; instead, it permits the more frequent performance of surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under 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 Condition for Operation have been performed within the specified time interval prior to entry into an applicable OPERATIONAL CONDITION or other specified applicability condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

APPLICABILITY

BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and addenda as required by 10 CFR 50.55a. 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 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable, and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined 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% delta k/k 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% delta k/k 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 specifications 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) the potential effects of the rod drop accident are limited. 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 so that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem; therefore, with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period that is 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 not fully 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 shut down for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in Section 15.4 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 the fuel cladding safety limit. 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

3/4.1.3 (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 before 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 6 inches in the event of a housing failure. The amount of rod reactivity that could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute 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 yet limited in frequency to avoid causing excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments that are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 20% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4 of the FSAR and the techniques of the analysis are presented in a topical report (Reference 1) and two supplements (References 2 and 3).

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high-power

CONTROL ROD PROGRAM CONTROLS

3/4.1.4 (Continued)

operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective, it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core and other piping systems connected to the reactor vessel. To allow for potential leakage and imperfect mixing, this concentration is increased by 20%. The required concentration is achieved by having a minimum available quantity of 4418 gallons of sodium-pentaborate solution containing a minimum of 5500 lb of sodium-pentaborate. This quantity of solution is a net amount which is above the pump suction, thus allowing for the portion that cannot be injected. The minimum pumping rate of 41.2 gpm per pump provides a negative reactivity insertion rate over the permissible penetaborate solution volume range, which adequately compensates for the positive reactivity effects from temperature and xenon during shutdown. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time when the system is inoperable or for longer periods of time when one of the redundant components is inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added; thus a check on the temperature and volume once every 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.

References:

1. C. J. Paone, R. C. Stirn, and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," GE 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.

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 an 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 limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 for two-recirculation-loop operation.

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared with previous analyses can be broken down as follows.

a. <u>Input Changes</u>

   1. Corrected vaporization calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

   2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

   3. Corrected guide tube thermal resistance.

   4. Correct heat capacity of reactor internals heat nodes.

b. <u>Model Change</u>

   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.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

3/4.2.1 (Continued)

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

- Break areas - The DBA break area was calculated more accurately.

b. Model Change

- 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 B3.2.1-1.

For plant operations with single recirculation loop the MAPLHGR limits of Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 are multiplied by 0.81. The constant factor 0.81, is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow-biased simulated thermal power-upscale scram setting and flow-biased neutron flux upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the fuel cladding integrity safety limit or that greater than or equal to 1% plastic strain does not occur in the degraded situation. The scram setpoint and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

BASES TABLE B3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS*

PARAMETERS	VALUE
Plant:	
1. Core THERMAL POWER	3461 MWt** which corresponds to 105% of rated steam flow
2. Vessel Steam Output	15.0×10^6 lbm/hr which corresponds to 105% of rated steam flow
3. Vessel Steam Dome Pressure	1055 psia
4. Design Basis Recirculation Line Break Area for:	
a. Large Breaks	3.1 ft^2
b. Small Breaks	0.09 ft^2

Fuel:

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.

† For single recirculation loop operation, loss of nucleate boiling is assumed at 0.1 second after LOCA regardless of initial MCPR.

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, 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 evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0.3 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154 (Reference 3) and the program used in non-pressurization events is described in NEDO-10802 (Reference 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 (Reference 4). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-2 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated. The K_f factors were derived using THERMAL POWER and core flow corresponding to 105% of rated steam flow.

The K_f factors were calculated so that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

MINIMUM CRITICAL POWER RATIO

3/4.2.3 (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, an MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts, while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures MCPR will be known following a change in THERMAL POWER or power shape, and therefore avoid operation while exceeding a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the linear heat generation rate (LHGR) in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation exceeding a thermal limit.

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.

BASES

References: (Continued)

3. Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154, October 1978.

4. TASC 01-A Computer Program for the Transient Analysis of a Single Channel, Technical Description, NEDE-25149, January 1980.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system (RPS) 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 maintenance is being performed. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter, and there are two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the

ISOLATION ACTUATION INSTRUMENTATION

3/4.3.2 (Continued)

high or low end of the setting has a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the FSAR Chapter 15 safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For AC-operated valves, it is assumed that the AC power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the DC-operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 13-second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for establishing emergency power 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 an allowance for instrument drift specifically allocated for each trip in the safety analysis. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

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 an allowance for instrument drift specifically allocated for each trip in the safety analysis. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

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, NEDO-24222, dated December 1979; and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system 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 from a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. When actuated, the EOC-RPT system trips both recirculation pumps to the low speed condition, thereby 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 switch which is administratively controlled by procedures. The manual bypasses and the automatic Operating Bypass at less than 30% 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., 190 milliseconds. Included in this time are: the time from initial valve movement to reaching the Trip Setpoint, the response time of the sensor, the response time of the system logic, and the time alloted for breaker arc suppression.

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

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

3/4.3.4 (Continued)

between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

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.

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 an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls, and Section 3/4.2, Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability. The scram discharge volume water level-high setpoint is referenced to a scram discharge volume instrument zero level at elevation 263 feet 10 inches.

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 50, Appendix A, General Design Criteria (GDC) 19, 41, 60, 61, 63, and 64.

MONITORING INSTRUMENTATION

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 ground motion effects 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 (RG) 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data are 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 RG 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 GDC 19 and 10 CFR 50.

The OPERABILITY of the remote shutdown system controls ensures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, controls and power circuits and transfer switches necessary to eliminate effects of a fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas in which a fire could damage systems normally used to shut down the reactor. This capability is consistent with GDC 3 and Appendix R to 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 RG 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants To Assess Plant Conditions During and Following an Accident," December 1980; and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

MONITORING INSTRUMENTATION

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors (SRMs) provide the operator with information about 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 being 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 (TIP) system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) before performing an LPRM function calibration. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core; thus, all detectors may not be required to be OPERABLE. The operability of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output with data obtained during the previous LPRM calibrations.

MONITORING INSTRUMENTATION

3/4.3.7.8 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 RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.9 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 before exceeding the limits of 10 CFR 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of GDC 60, 63, and 64 of Appendix A to 10 CFR 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

3/4.3.7.10 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 before exceeding the limits of 10 CFR 20. The range of the noble gas channels of the main stack and radwaste/reactor building vent effluent monitors is sufficiently large to envelope both normal and accident levels of noble gas activity. The capabilities of these instruments are 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 1980 and NUREG-0737, "Clarification of the TMI Action Plan Requirements," November 1980. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the offgas system. The OPERABILITY and use of this instrumentation is consistent with the requirements of GDC 60, 63, and 64 of Appendix A to 10 CFR 50.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

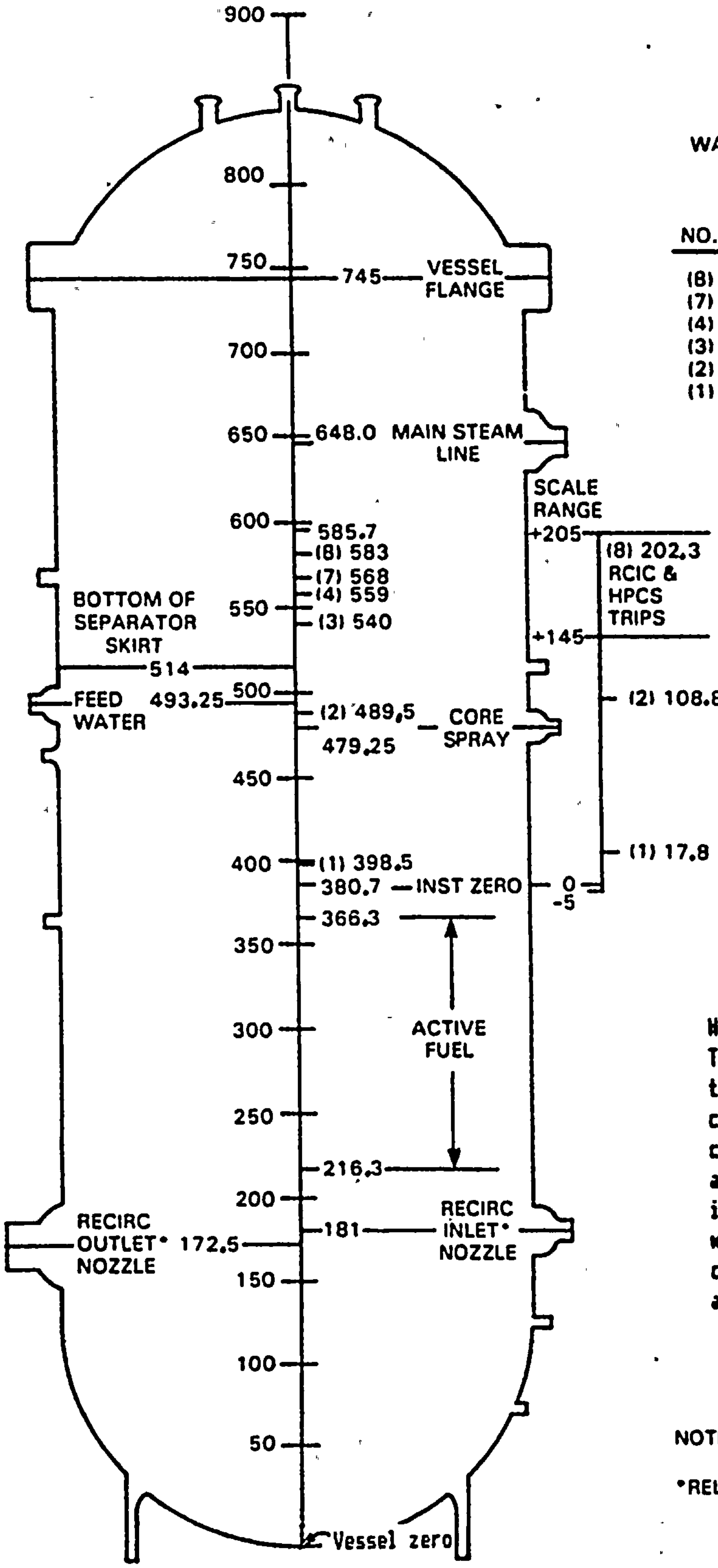
This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and

TURBINE OVERSPEED PROTECTION SYSTEM (cont'd)

will protect the turbine from excessive overspeed. Protection from excessive turbine overspeed is required since excessive overspeed could generate potentially damaging missiles which could impact and damage safety-related components, equipment, or structures.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

The plant systems actuation instrumentation is provided: (1) to initiate action of the feedwater system/main turbine trip system in the event of feedwater controller failure and (2) to ensure the proper operation of the service water system during normal and accident conditions.



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (in.)	READING
(8)	583	202.3
(7)	568	187.3
(4)	559	178.3
(3)	540	159.3
(2)	489.5	108.8
(1)	398.5	17.8

WIDE RANGE LEVEL
This indication is coolant temperature sensitive. The calibration is made at rated conditions. The level error at low pressures (temperatures) is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap and not indicated level.

NOTE: DIMENSIONS IN INCHES

*RELATIVE TO VESSEL

Bases Figure B3/4.3-1. Reactor Vessel Water Level

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively, MAPLHGR limits are decreased by the factor given in Specification 3.2.1, and MCPR operating limits are adjusted per Section 3/4.2.3.

Additionally, surveillance on the volumetric flow rate of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 30%* THERMAL POWER or 50%* rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during the extended operation of the single recirculation loop mode.

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high-power/low-flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists, depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.6 was chosen as the basis for determining the generic region for surveillance to account for the plant-to-plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.

Plant-specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case, the degree of conservatism can be reduced since plant-to-plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron

*Initial values. The final values are determined during startup testing based upon the threshold THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head, preventing stratification.

RECIRCULATION SYSTEM

3/4.4.1 (Continued)

flux noise levels between 1% and 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability-related neutron flux limit cycle oscillations occur, they result in peak-to-peak neutron flux limit cycles of 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specified core flow can be applied over a range of core flows. To maintain a reasonable variation between the low-flow and high-flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline data taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

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 shutting down the facility when a jet pump is 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 the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop after a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

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 before 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 SYSTEM

3/4.4.1 (Continued)

recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference $\geq$ 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

3/4.4.2 SAFETY/RELIEF VALVES

The safety/relief valves operate during a postulated ATWS event to prevent the reactor coolant system from being pressurized above a design allowable value of 1375 psig in accordance with the ASME Code. A total of 16 OPERABLE safety/relief valves is required to limit local pressure at active components to within ASME III allowable design values (Service Level A). All other appropriate ASME III limits are also bounded by this requirement.

The safety-relief valve lift settings will be demonstrated only during shutdown in accordance with the provisions of Specification 4.0.5.

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 RG 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 background leakage normally expected to result from equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2-ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present, so a 0.5-ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole-body doses resulting from a main steam line failure outside the containment during steady-state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomena that may occur following changes in THERMAL POWER.

Closing the main steam line isolation valves limits 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 from temperature and pressure changes in the system. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 are derived from the fracture toughness requirements of 10 CFR 50, Appendix G, and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Subsection 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Bases Table B3/4.4.6-1. 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 B3/4.4.6-1 and the recommendations of RG 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating irradiated specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3 shall be adjusted, as required, on the basis of the specimen data and recommendations of RG 1.99, Revision 1. Data determined from specimens removed at the end of the first cycle will be used to adjust the fluence of Bases Figure B3/4.4.6-1.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1 and 3.4.6.1-3, for inservice hydrostatic testing and leak testing and for reactor criticality have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR 50.

BASES TABLE B3/4.4.6-1

LIMITING REACTOR VESSEL TOUGHNESS

COMPONENT	WELD SEAM ID OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU(%)	P(%)	STARTING RT_{NDT} (°F)	MAX.* ΔRT_{NDT} (°F)	UNIRRADIATED UPPER SHELF (FT-LB)	MAX. RT_{NDT} (°F)
Beltline								
Plate	SA-533, Gr. B, Cl. 1	C3147-2	0.11	0.012	0	30	88	+30
Weld	Seams BA, BB, & BC	5P6214B/0331	0.014	0.011	-40	18	97	-22
Non-Beltline								
Shell Ring	SA-533, Gr. B, Cl. 1	All Plates						+10
Bottom Head Dome	SA-533, Gr. B, Cl. 1	C3073/2						+10
Bottom Head Torus	SA-533, Gr. B, Cl. 1	C3073/2						+10
Top Head Dome	SA-533, Gr. B, Cl. 1	A0678/1						-20
Top Head Torus	SA-533, Gr. B, Cl. 1	C2325/2						-1
Top Head Flange	SA-508, Cl. 2	49D161, 49B168						-30
Vessel Flange	SA-508, Cl. 2	48D1072, 48B1121						-20
LPCI Nozzle**	SA-508, Cl. 2	Q2QL3W						-20
Feedwater Nozzle	SA-508, Cl. 2	Q2QL2W						-20
Weld	INMM/LINDE 124	All Heats						-20
Closure Studs	SA-540, Gr. B24	All Heats						+10†

* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}.

** The design location of the low-pressure core injection (LPCI) nozzles results in these components and their related vessel welds to experience and end-of-life (EOL) fluence of 1.7×10^{17} n/cm² (E>1 MeV). As a result, the nozzles are predicted to have an EOL RT_{NDT} of -13°F and the limiting weld material will have an EOL RT_{NDT} of -12°F.

† Meet 45 ft-lb and 25 mils lateral expansion requirement at 10°F.

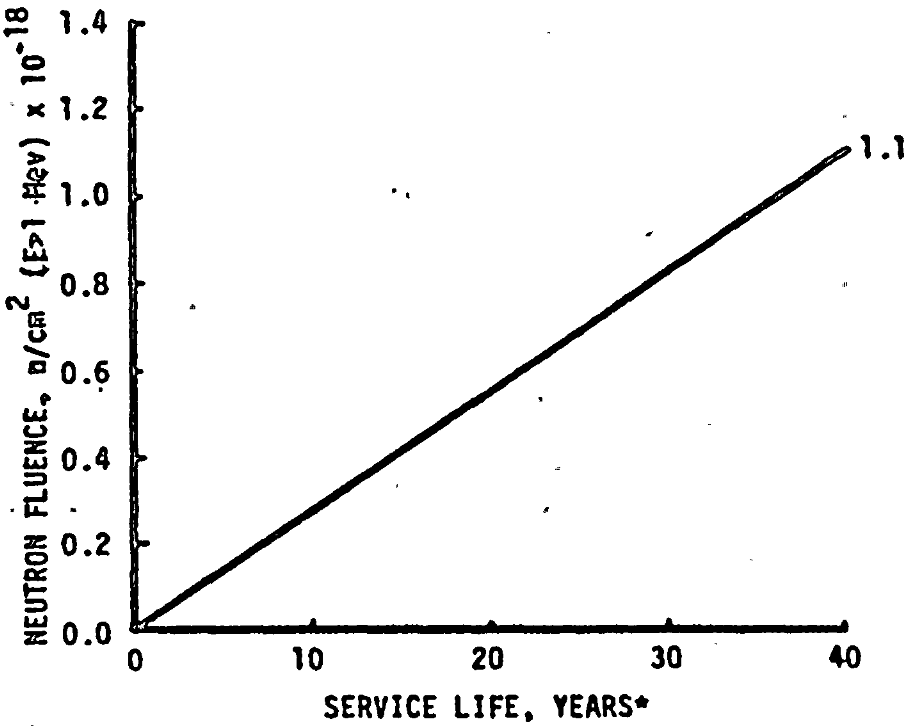


FIGURE B 3/4 4.6-1 FAST NEUTRON FLUENCE (E>1 Mev) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

*At 90% of RATED THERMAL POWER and 90% availability.

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 of 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, 1980 Edition, and Addenda through Winter of 1980.

The inservice inspection program for ASME Code Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication; however, single-failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

3/4.5 EMERGENCY CORE COOLING SYSTEM

BASES

3/4.5.1 & 3/4.5.2 ECCS - OPERATING AND SHUTDOWN

ECCS Division I consists of the low-pressure core spray system and low-pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A." ECCS Division II consists of low-pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B."

The low-pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and 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 is a source for flooding 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 the reactor to be shut down. The pump discharge piping is maintained full to prevent water hammer damage to piping.

The low-pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The Surveillance Requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires the reactor to be shut down. The pump discharge piping is maintained full to prevent water hammer damage to piping.

ECCS Division III consists of the high-pressure core spray (HPCS) system. The HPCS system is provided to assure that the reactor core is adequately cooled to limit fuel cladding temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1160 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

ECCS - OPERATING AND SHUTDOWN

3/4.5.1 & 3/4.5.2 (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 516/1550/6350 gpm at differential pressures of 1160/1130/200 psi, respectively. Initially, water from the condensate storage tank is used instead of water injected 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, a system for which no credit is taken in the safety analysis, will automatically provide makeup water 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 the reactor to be shut down. The pump discharge piping is maintained full to prevent water hammer damage.

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 five valves. It is, therefore, appropriate to permit two valves to be out of service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION POOL

The suppression 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 3.6.2.1.

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 equal to that required for OPERATIONAL CONDITIONS 1, 2, and 3 because in all cases the minimum water volume is based on NPSH, recirculation volume, and vortex prevention in accordance with NUREG-0869 (April 1983, issued for comment).

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the control room and site boundary radiation doses to within the limits of General Design Criterion (GDC) 19 and 10 CFR 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 39.75 psig, Pa. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 La during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore, the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening. Leak testing of valves in potential bypass leakage pathways is performed at a test pressure of 40.00 psig rather than Pa, 39.75 psig, for consistency with the accident analysis.

The leakage rates specified for the main steam line isolation valves, the main steam drain line isolation valves, and the postaccident sampling system gas sample and return line block valves are used to quantify the maximum amount of primary containment atmosphere that could bypass secondary containment and leak directly to the environment after a design-basis loss-of-coolant accident. These data are used to determine the radiological consequences of this accident and ensure that the resultant doses are within the limits of GDC 19 and 10 CFR 100.

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.

PRIMARY CONTAINMENT

3/4.6.1.4 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 design pressure of 45 psig in the event of a loss-of-coolant accident (LOCA). A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.5 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 39.75 psig does not exceed the design pressure of 45.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 4.7 psi. The limit of 14.2 to 15.45 psia for initial positive containment pressure will limit the total pressure to 39.75 psig, which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.6 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

In addition, the maximum drywell average air temperature is also the limiting initial condition used to determine the maximum negative differential pressure acting on the drywell and suppression chamber following inadvertent actuation of the containment sprays.

3/4.6.1.7 PRIMARY CONTAINMENT PURGE SYSTEM

The 14-inch drywell and 12-inch suppression chamber supply and exhaust valves are limited to 90 hours of use per 365 days during purge or vent operations in OPERATIONAL CONDITIONS 1, 2, and 3 to meet the requirements of Branch Technical Position CSB 6-4 for valves greater than 8 inches in diameter. The requirement to limit the opening of 2CPS*AOV105, 2CPS*AOV107, 2CPS*AOV109, and 2CPS*AOV110 to 70 degrees, and 2CPS*AOV111 to 60 degrees ensures these valves will close during a LOCA or steam line break accident, and therefore, the site boundary dose guidelines of 10 CFR 100 would not be exceeded in the event of an accident during purging or venting operations.

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PURGE SYSTEM

3/4.6.1.7 (Continued)

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 leakage limit shall not be exceeded when the leakage rates are determined to be less than or equal to 4.38 scf/hour per 14-inch valve and 3.75 scf/hour per 12-inch valve when pressurized to 39.75 or 40.0 psig, as applicable.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression pool water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression pool water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Because all of the gases in the drywell are purged into the suppression pool air space during a LOCA, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design-basis accident is approximately 40 psig, which is below the design pressure of 45 psig. Maximum water volume of 154,794 cubic feet results in a downcomer submergence of 11 feet 0 inch, and the minimum volume of 145,495 cubic feet results in a submergence approximately 18 inches less. The majority of the Bodega Bay tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as detailed in Specification 3.5.3.

PRIMARY CONTAINMENT

DEPRESSURIZATION SYSTEMS

3/4.6.2 (Continued)

Under full-power operating conditions, blowdown to the suppression pool at the initial water temperature of 90°F results in a water temperature of approximately 140°F immediately following blowdown which is below the 200°F used for complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head (NPSH) exceeds that required by both the residual heat removal (RHR) and core spray pumps; thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations. For purposes of temperature monitoring, the suppression pool is divided into 10 sectors. Temperature elements in each sector ensure adequate monitoring of pool water temperature in the area of the main steam safety/relief valve discharge quenchers.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly, and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum, this action shall include: (1) use 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, separate their discharge from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of

PRIMARY CONTAINMENT

PRIMARY CONTAINMENT ISOLATION VALVES

3/4.6.3 (Continued)

GDC 54 through 57 of Appendix A to 10 CFR 50. Measurement of the closure time of automatic containment isolation valves is performed for the purpose of demonstrating PRIMARY CONTAINMENT INTEGRITY and system OPERABILITY (Specification 3/4.6.1).

The maximum isolation times for primary containment automatic isolation valves listed in this specification are either the analytical times used in the accident analysis as described in the FSAR; or times derived by applying margins to the vendor test data obtained in accordance with industry codes and standards. For non-analytical automatic primary containment isolation valves, the maximum isolation time is derived as follows:

1) Valves with full stroke times less than or equal to 10 seconds, maximum isolation time approximately equals the vendor tested closure time multiplied by 2.0.

2) Valves with full stroke time greater than 10 seconds, maximum isolation time approximately equals the vendor tested closure time multiplied by 1.5. Valve closing times do not include isolation instrumentation response times.

Valve closing times do not include isolation instrumentation response times.

3/4.6.4 SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four pairs of valves to provide redundancy so that operation may continue for up to 72 hours with no more than one pair of vacuum breakers inoperable in the closed position.

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times, the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a subatmospheric condition in the reactor building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

PRIMARY CONTAINMENT

SECONDARY CONTAINMENT

3/4.6.5 (Continued)

The drawdown time limit has been established considering the same fan performance and buildng inleakage assumptions as in the post-LOCA analysis except that, since the surveillance test is performed when the plant is shut down, (1) post-LOCA heat-loads are not present; (2) the initial secondary containment pressure is atmospheric; and (3) loss of offsite power is not assumed. Meeting this drawdown time verified that secondary containment leakage and fan performance are consistent with the assumptions of the LOCA analysis.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters operating for 10 hours during each 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and high-efficiency particulate air (HEPA) filters.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the primary containment below its flammable limit during post-LOCA conditions. The drywell and suppression chamber hydrogen recombiner system is capable of controlling the expected hydrogen and oxygen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The hydrogen control system is consistent with the recommendations of RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 PLANT SERVICE WATER SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal or 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.

Independence in the plant service water system, as required by the specification, is achieved by OPERABILITY of the divisional separation logic and valves (2SWP*MOV50A, 2SWP*MOV50B). During normal plant operating conditions, the two divisions of the service water system are interconnected. The intake deicing heater specification ensures that adequate water is available to the service water system. In order to prove that the system is supplying adequate heat to the bar racks, a portable ammeter shall be used to check the full load current of the heaters. The current should be checked on a weekly basis. Current shall be measured for each phase at each of the four motor control center locations. If a major deviation from rated current is detected, further investigation is required to determine if an open circuit exists in the individual heater circuits.

The 18-month check of circuit readings will check against long-term degradation of circuit insulations.

3/4.7.2 REVETMENT-DITCH STRUCTURE

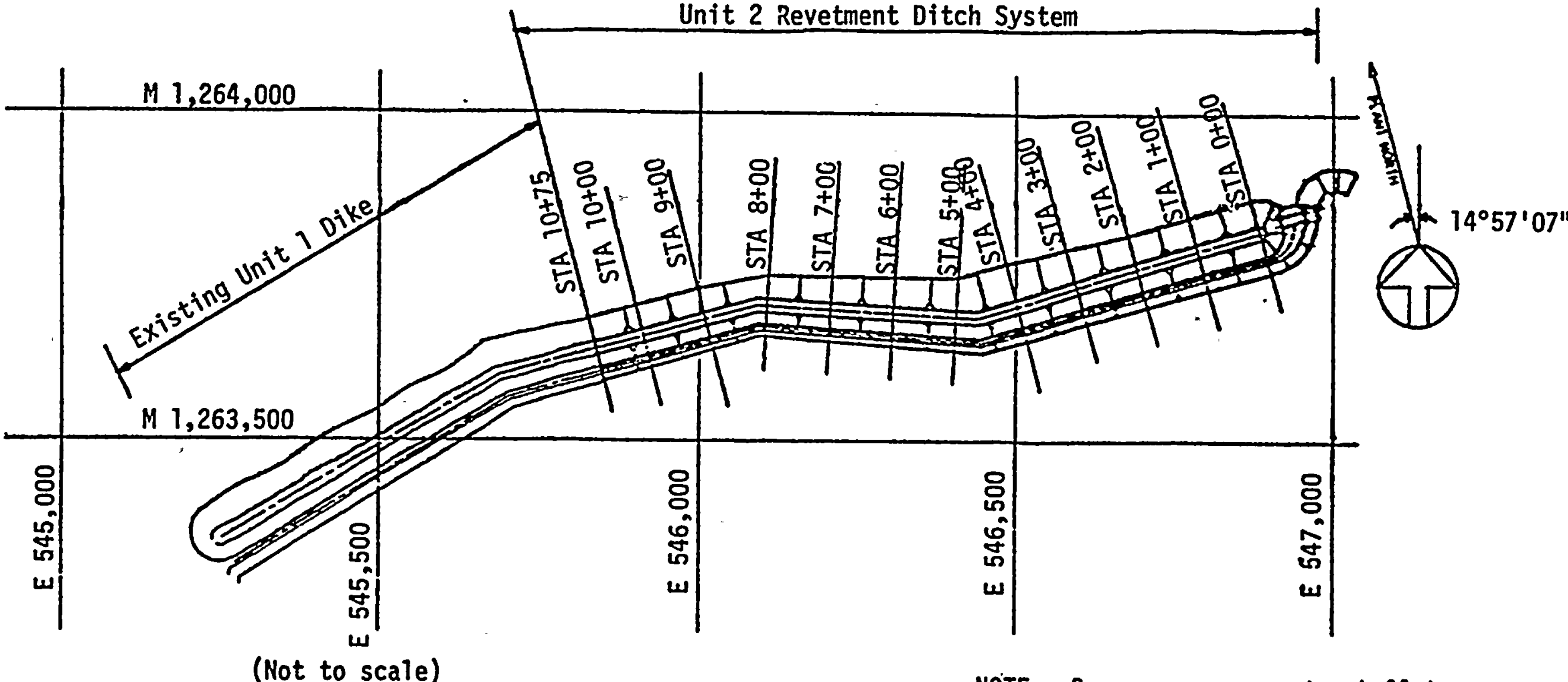
The purpose of the revetment-ditch structure is to protect the plant fill and foundation from wave erosion, expected during the probable maximum windstorm for a maximum still water elevation of 254 feet.

The revetment-ditch structure is Seismic Category I and is designed to withstand the impact of waves. So long as the fill is in place, waves cannot impact Category I structures because of the lack of sufficient depth of water to sustain such waves.

The revetment-ditch structure can sustain a high degree of damage and still perform its function, protecting the site fill from erosion. Thus, the operability condition for operation of the revetment-ditch structure has been written to ensure that severe damage to the structure will not go undetected for a substantial period of time and to provide for prompt corrective action and NRC notification.

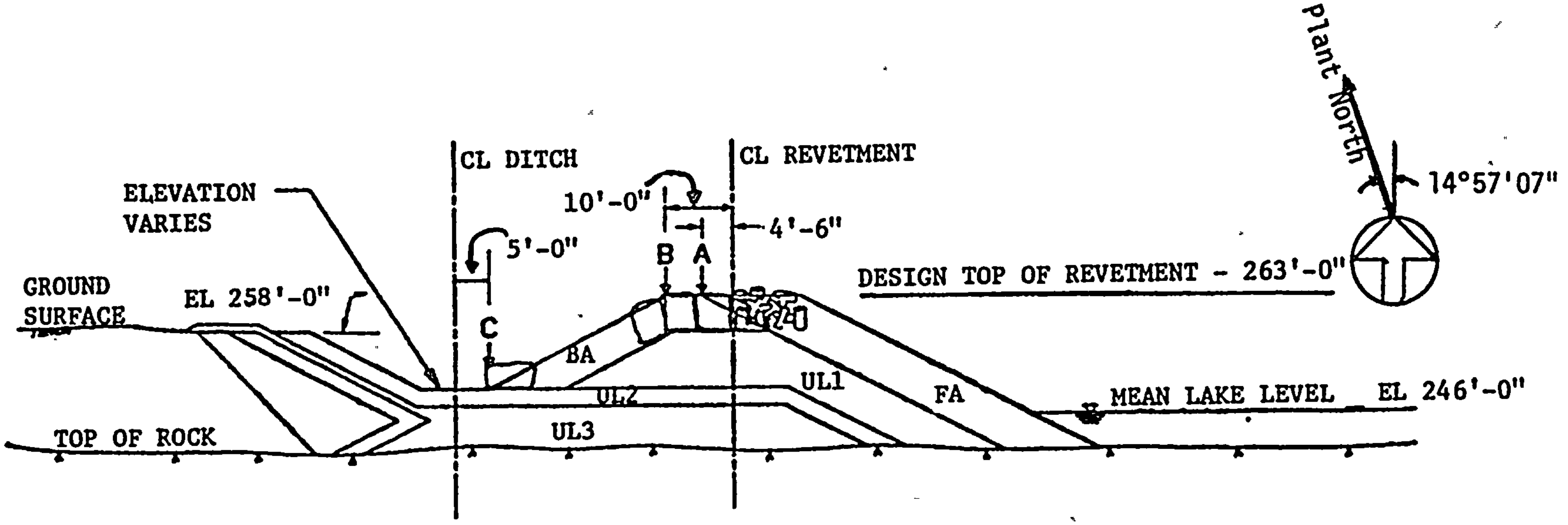
3/4.7.3 CONTROL ROOM OUTDOOR AIR SPECIAL FILTER TRAIN SYSTEM

The OPERABILITY of the control room outdoor air special filter train 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



NOTE: Survey measurements shall be taken at 100 ft stations as shown. All station locations are approximate and will be determined upon installation of survey mounts.

Bases Figure B3/4.7.2-1 Plan View - Revetment-Ditch Structure, Inservice Inspection Station Locations



LEGEND

FA	Front Armor, Double Layer of 4900 lb Dolids Units
UL1	First Underlayer, 2000 to5000 lb Stone Units
UL2	Second Underlayer, 75 to 250 lb Stone Units
UL3	Third Underlayer, 2.3 to 15 lb Stone Units
BA	Back Armor, Single Layer of 10,000 to 16,000 lb Stone Armor Units
A,B,C	Survey Points

Bases Figure B3/4.7.2-2 Typical Section - Revetment-Ditch Structure, Inservice Inspection Station Locations

CONTROL ROOM OUTDOOR AIR SPECIAL FILTER TRAIN SYSTEM

3/4.7.3 (Continued)

each 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and high-efficiency particulate air (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 GDC 19 of Appendix A to 10 CFR 50.

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the emergency core cooling system (ECCS) 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 RCIC system 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.

The Surveillance Requirements provide adequate assurance that RCIC will be OPERABLE when required. All active components are testable and full flow can be demonstrated by recirculation during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage.

3/4.7.5 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event that initiates dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed

SNUBBERS

3/4.7.5 (Continued)

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 originally 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 that 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.

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 $35(1 + c/2)$ 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.

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.

3/4.7.6 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium.

This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, and surveillance requirements are commensurate with the probability of damage to a source in that group. Those sources that are frequently handled are required to be tested more often than those that are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation-monitoring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.7 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis of FSAR Chapter 15.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2, & 3/4.8.3 AC SOURCES, DC SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the AC and DC 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 AC and DC power sources and distribution systems satisfy the requirements of GDC 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 consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least Division I or II of the onsite AC and DC 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 AC or DC source. Division III supplies the high-pressure core spray (HPCS) system only.

The AC and DC source allowable out-of-service times are based on RG 1.93, "Availability of Electrical Power Sources," December 1974. When diesel generator EDG*1 (Division I) or EDG*3 (Division II) 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 EDG*1 or EDG*3 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 EDG*1 or EDG*3 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 AC and DC 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 RG 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; RG 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and RG 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

AC SOURCES, DC SOURCES, AND ONSITE POWER DISTRIBUTION SYSTEMS

3/4.8.1-3 (Continued)

The Surveillance Requirements for demonstrating the OPERABILITY of the unit batteries are in accordance with the recommendations of RG 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Standard 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8.2.1-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full-charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full-charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full-charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8.2.1-1 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full-charge specific gravity ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 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

Primary containment electrical penetrations and penetration conductors are protected by either de-energizing circuits not required during reactor operation or demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers by periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provides assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The emergency lighting system overcurrent protective devices ensure that a failure of the non-Class 1E portion of the circuit will not affect the operation of the remaining portions of the Class 1E circuits that are necessary for safe shutdown.

The EPAs provide Class 1E isolation capabilities for the RPS power supplies and the scram power supplies. This is required because the power supplies are not Class 1E power supplies.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during other CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality before fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

BASES

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 & 3/4.9.9 WATER LEVEL, REACTOR VESSEL AND WATER LEVEL, AND SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING and (2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 22 feet 3 inches of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 22 feet 3 inches of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low-power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the Technical Specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirments ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this Limiting Condition for Operation.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no-flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access, the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize the discharge of contaminated water to the radioactive waste disposal system.

3/4.10.7 SPECIAL INSTRUMENTATION - INITIAL CORE LOADING

This special test exception permits relief from the requirements for a minimum count rate while loading the first 16 fuel bundles to allow sufficient source-to-detector coupling such that minimum count rate can be achieved on an SRM. This is acceptable because of the significant margin to criticality while loading the initial 16 fuel bundles.

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 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 to 10 CFR 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR 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.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

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 L. A. Currie, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

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 to 10 CFR 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 potable drinking water that are in excess of the requirements of 40 CFR 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, so 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 that result from actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in RG 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,"

LIQUID EFFLUENTS

DOSE

3/4.11.1.2 (Continued)

Revision 1, October 1977 and R.G. 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment before release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept as low as is reasonably achievable. This specification implements the requirements of 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50 and the design objective given in Section II.D of Appendix I to 10 CFR 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 to 10 CFR 50 for liquid effluents. This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose rate at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR 20 to UNRESTRICTED AREAS.

GASEOUS EFFLUENTS

DOSE RATE

3/4.11.2.1 (Continued)

The annual dose limits are the doses associated with the concentrations of 10 CFR 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 20.106(b). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the whole body or to less than or equal to 3000 mrem/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 mrem/year. This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in L. A. Currie, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environments Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

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 to 10 CFR 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 guidelines of Appendix I be shown by calculational procedures based on models and data so 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 from the actual release rates of radioactive noble gases

GASEOUS EFFLUENTS

DOSE - NOBLE GASES

3/4.11.2.2 (Continued)

in gaseous effluents are consistent with the methodology provided in RG 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 RG 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 or beyond the SITE BOUNDARY are based upon real-time meteorological conditions or the historical average atmospheric conditions. This specification applies to the release of radioactive material in gaseous effluents from each unit at the site.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I to 10 CFR 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, so 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 from the actual release rates of the subject materials are consistent with the methodology provided in RG 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 RG 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 radioactive material in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at or beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radioactive material, (2) deposition of radioactive material onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk-producing animals and meat-producing animals graze (human consumption of the

BASES

GASEOUS EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

3/4.11.2.3 (Continued)

milk and meat is assumed), and (4) deposition on the ground with subsequent exposure to man. This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.4 & 3/4.11.2.5 GASEOUS RADWASTE TREATMENT SYSTEM AND VENTILATION EXHAUST TREATMENT SYSTEM

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment before 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 50.36a, GDC 60 of Appendix A to 10 CFR 50, and the design objectives given in Section II.D of Appendix I to 10 CFR 50. Limits governing the use of appropriate portions of the system were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I to 10 CFR 50, for gaseous effluents. This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportional among the units sharing that system.

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 GASEOUS RADWASTE TREATMENT SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen concentrations from reaching these flammability limits. These automatic control features include injection of dilutants to reduce concentrations below flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of GDC 60 of Appendix A to 10 CFR 50.

3/4.11.2.7 MAIN CONDENSER - OFFGAS

Restricting the gross radioactivity rate of noble gases from the main condenser offgas 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 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of GDC 60 and 64 of Appendix A to 10 CFR 50.

GASEOUS EFFLUENTS

3/4.11.2.8 VENTING OR PURGING

This specification provides reasonable assurance that releases from drywell and/or suppression chamber purging operations will not exceed the annual dose limits of 10 CFR 20 for unrestricted areas.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and GDC 60 of Appendix A to 10 CFR 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 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from releases of radioactivity and from radiation from uranium fuel cycle sources exceed 25 mrem to the whole body or any organ, except the thyroid (which shall be limited to less than or equal to 75 mrem). For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units including outside storage tanks, etc., are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR 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 5 miles must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR 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 the individual 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 exposure of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. After 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 a before-the-fact limit representing the capability of a measurement system and not as an after-the-fact limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in L. A. Currie, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at or beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. The best information, such as from a door-to-door survey, from an 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 50. Restricting the census to gardens of greater than 500 square feet 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 RG 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 vegatation (i.e., similar to lettuce and cabbage) and (2) the vegetation yield was 2 kg/m^2.

A MILK SAMPLING LOCATION, as defined in Section 1.0, requires that at least 10 milking cows are present at a designated milk sample location. It has been

BASES

LAND USE CENSUS

3/4.12.2 (Continued)

found from past experience, and as a result of conferring with local farmers, that a minimum of 10 milking cows is necessary to guarantee an adequate supply of milk twice a month for analytical purposes. Locations with fewer than 10 milking cows are usually utilized for breeding purposes, eliminating a stable supply of milk for samples as a result of suckling calves and periods when the adult animals are dry.

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 materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR 50.

5.0 DESIGN FEATURES

5.1 SITE

The Nine Mile Point Nuclear Station and James A. FitzPatrick Nuclear Power Plant site comprising approximately 1500 acres, is located on the shores of Lake Ontario, about 7 miles northeast of Oswego, New York. An exclusion distance of nearly 4600 feet is provided between the station and the nearest SITE BOUNDARY to the west, a mile to the boundary on the east, and a mile and a half to the southern SITE BOUNDARY.

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.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1.3-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel-lined concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined concrete vessel in the shape of a truncated cone on top of a water-filled suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 303,418 cubic feet. The suppression chamber has a minimum air region of 192,028 cubic feet and a minimum water region of 145,495 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure, 45 psig.

b. Maximum internal temperature:

- drywell, 340°F
- suppression pool, 212°F
- suppression chamber, 270°F

CONTAINMENT

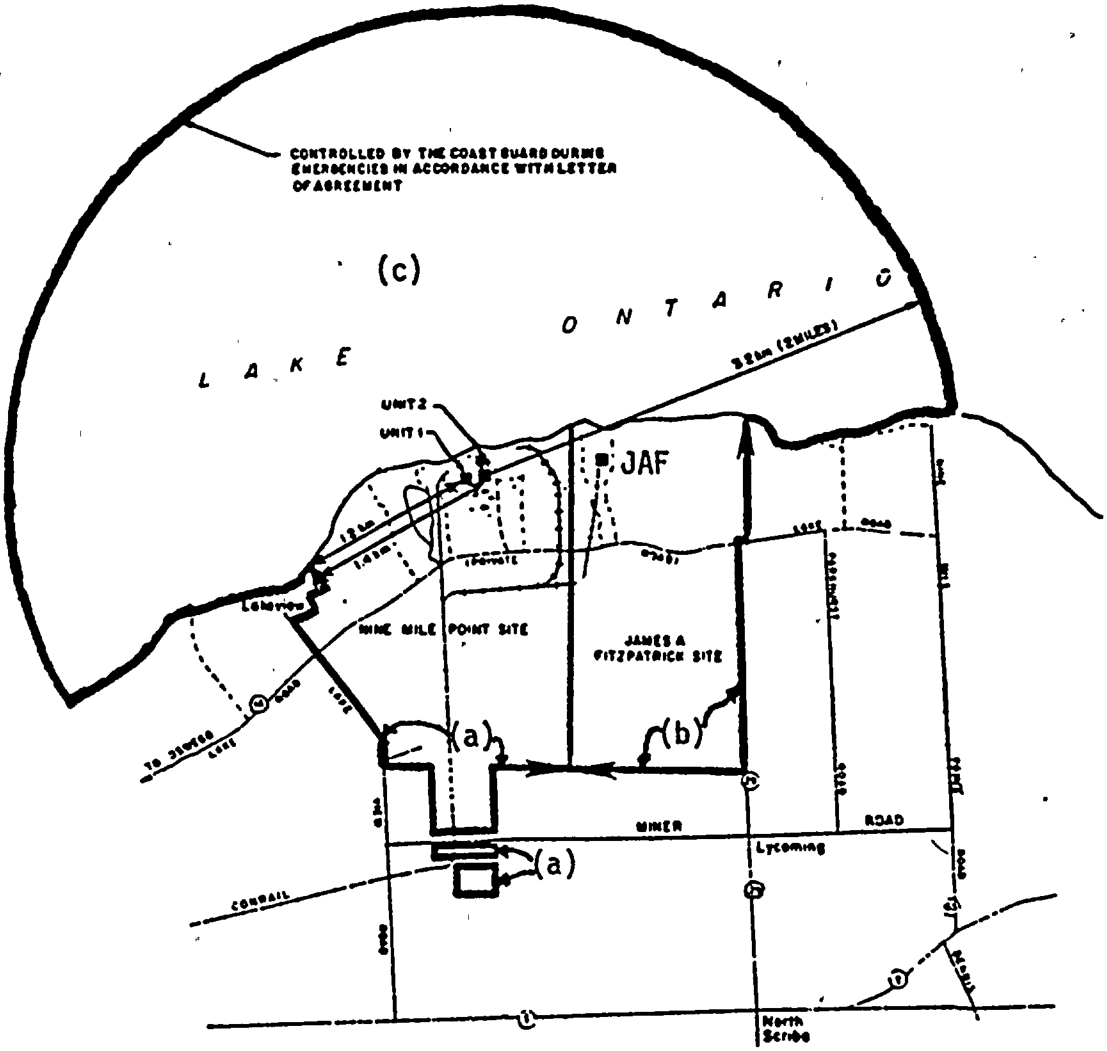
DESIGN TEMPERATURE AND PRESSURE

5.2.2 (Continued)

c. Maximum external pressure, 4.7 psig.

d. Maximum floor differential pressure:

- 25 psid, downward.
- 10 psid, upward.



EXCLUSION AREA BOUNDARY

(a) Niagara Mohawk Power Corp. Property Line
(b) New York Power Authority Property Line
(c) Region Controlled By Coast Guard During Emergencies.

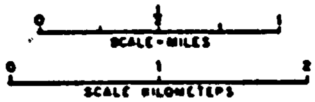


Figure 5.1.1-1 Exclusion Area Boundary

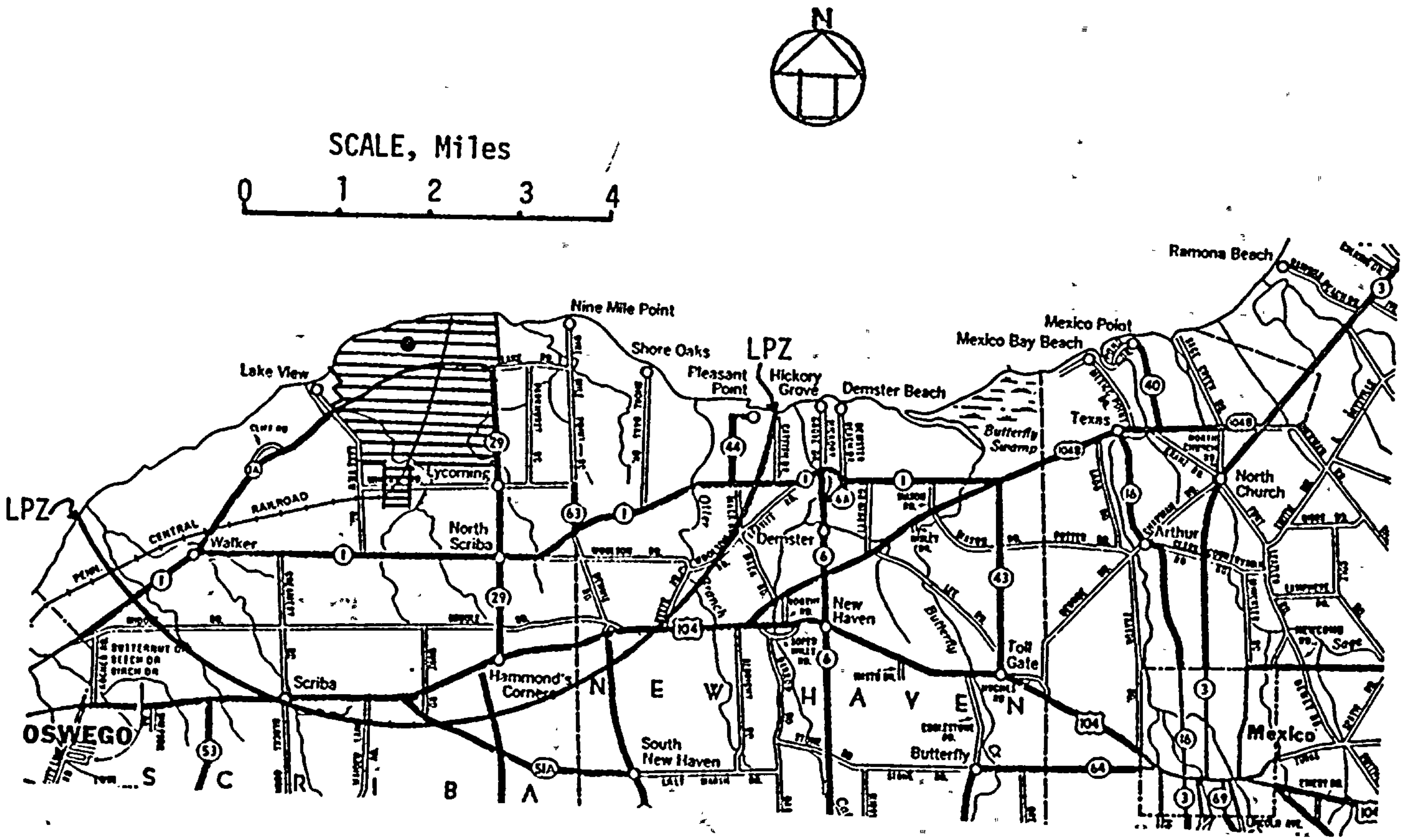


Figure 5.1.2-1 Low Population Zone

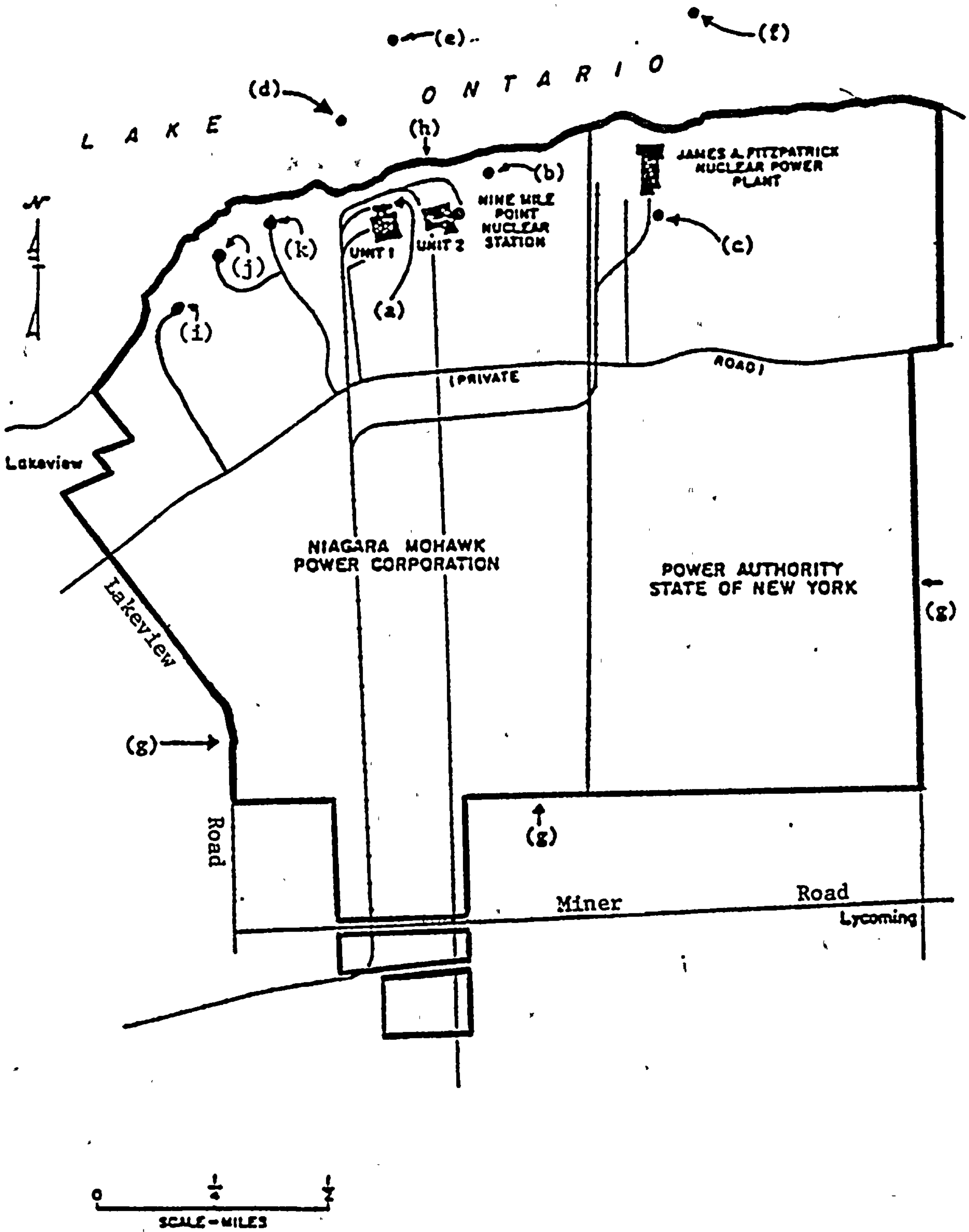


Figure 5.1.3-1 Site Boundaries

NOTES TO FIGURE 5.1.3-1

(a) NMP1 Stack (height is 350')
(b) NMP2 Stack (height is 430')
(c) JAFNPP Stack (height is 385')
(d) NMP1 Radioactive Liquid Discharge (Lake Ontario, bottom)
(e) NMP2 Radioactive Liquid Discharge (Lake Ontario, bottom)
(f) JAFNPP Radioactive Liquid Discharge (Lake Ontario, bottom)
(g) Site Boundary
(h) Lake Ontario Shoreline
(i) Meteorological Tower
(j) Training Center
(k) Energy Information Center

Additional Information:

- NMP2 Reactor Building Vent is located 187 feet above ground level
- JAFNPP Reactor and Turbine Building Vents are located 173 feet above ground level
- JAFNPP Radwaste Building Vent is 112 feet above ground level
- The Energy Information Center and adjoining picnic area are UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC
- Lake Road, a private road, is an UNRESTRICTED AREA within the SITE BOUNDARY accessible to MEMBERS OF THE PUBLIC

DESIGN FEATURES

CONTAINMENT

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the reactor building, and the north and south auxiliary bays and has a minimum free volume of 3,876,630 cubic feet.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies; each fuel assembly will contain 62 fuel rods and two water rods clad with Zircaloy-2. Each fuel rod shall have a nominal active fuel length of 150 inches. The initial core loading shall have a maximum average enrichment of 1.88 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing 143 inches of boron carbide, B_4C, powder surrounded by a cruciform-shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements

b. For a pressure of:

   1. 1250 psig on the suction side of the recirculation pump

   2. 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 22,000 cubic feet at a nominal steam dome saturation (average) temperature of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION

The meteorological tower shall be located as shown on Figure 5.1.3-1.

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 of the FSAR.

b. A nominal 6.18-inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel stored in the new fuel storage racks shall not exceed 0.95 in the normal dry condition or in the abnormal completely water-flooded condition. The k_{eff} shall not exceed 0.98 with all but one of the non-combustible storage vault covers in place when optimum moderation (foam, spray, fogging, or small droplets) is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 329 ft. 7 in.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4049 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

The component identified in Table 5.7.1-1 is designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

TABLE 5.7.1-1

REACTOR CYCLIC OR TRANSIENT LIMITS AND DESIGN CYCLE OR TRANSIENT

CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
120 heatup and cooldown cycles	70°F to 565°F to 70°F
80 step change cycles	Loss of feedwater heaters
198 reactor trip cycles	100% to 0% of RATED THERMAL POWER
130 hydrostatic pressure and leak tests	Pressurized to $\geq$930 psig and $\leq$1250 psig

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Superintendent - Nuclear Generation shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during the Superintendent's absence.

6.1.2 The Station Shift Supervisor - Nuclear (or during the Supervisor's absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President - Nuclear Generation shall be reissued to all station personnel annually.

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 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 OPERATIONAL CONDITIONS 1, 2, or 3, at least one Licensed Senior Operator or Licensed Operator shall be at the controls of the unit.

c. A Radiation Protection Technician* shall be on site when fuel is in the reactor;

d. At least two Licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown, and during recovery from reactor trips.

e. A Licensed Senior Operator shall be required in the control room during OPERATIONAL CONDITIONS 1, 2, and 3 and when the emergency plan is activated. This may be the Station Shift Supervisor - Nuclear, the Assistant Station Shift Supervisor - Nuclear or other individuals with a valid senior operator license. When the emergency plan is activated in OPERATIONAL CONDITIONS 1, 2, or 3 the Assistant Station Shift Supervisor - Nuclear becomes the Shift Technical Advisor and the Station Shift

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* 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. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming crewman being late or absent.

ORGANIZATION

UNIT STAFF

6.2.2.e (Continued)

Supervisor - Nuclear is restricted in the control room until an additional Licensed Senior Operator arrives.

f. A Licensed Senior Operator shall be responsible for all movement of new and irradiated fuel within the site boundary. All core alterations shall be directly supervised by a Licensed Senior Operator who has no other concurrent responsibilities during this operation. A Licensed Operator will be required to manipulate the controls of all fuel handling equipment except movement of new fuel from receipt through dry storage. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.

g. A Fire Brigade* of five members shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and 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.

h. 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.

i. 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 7-day period, all excluding shift turnover time.

   3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

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* The radiation protection qualified individual 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.

ORGANIZATION

UNIT STAFF

6.2.2.i (Continued)

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 General Superintendent - Nuclear Generation, 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 so that individual overtime shall be reviewed monthly by the General Superintendent - Nuclear Generation or a designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP

FUNCTION

6.2.3.1 The Independent Safety Engineering Group (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 Supervisor Technical Support - Nuclear.

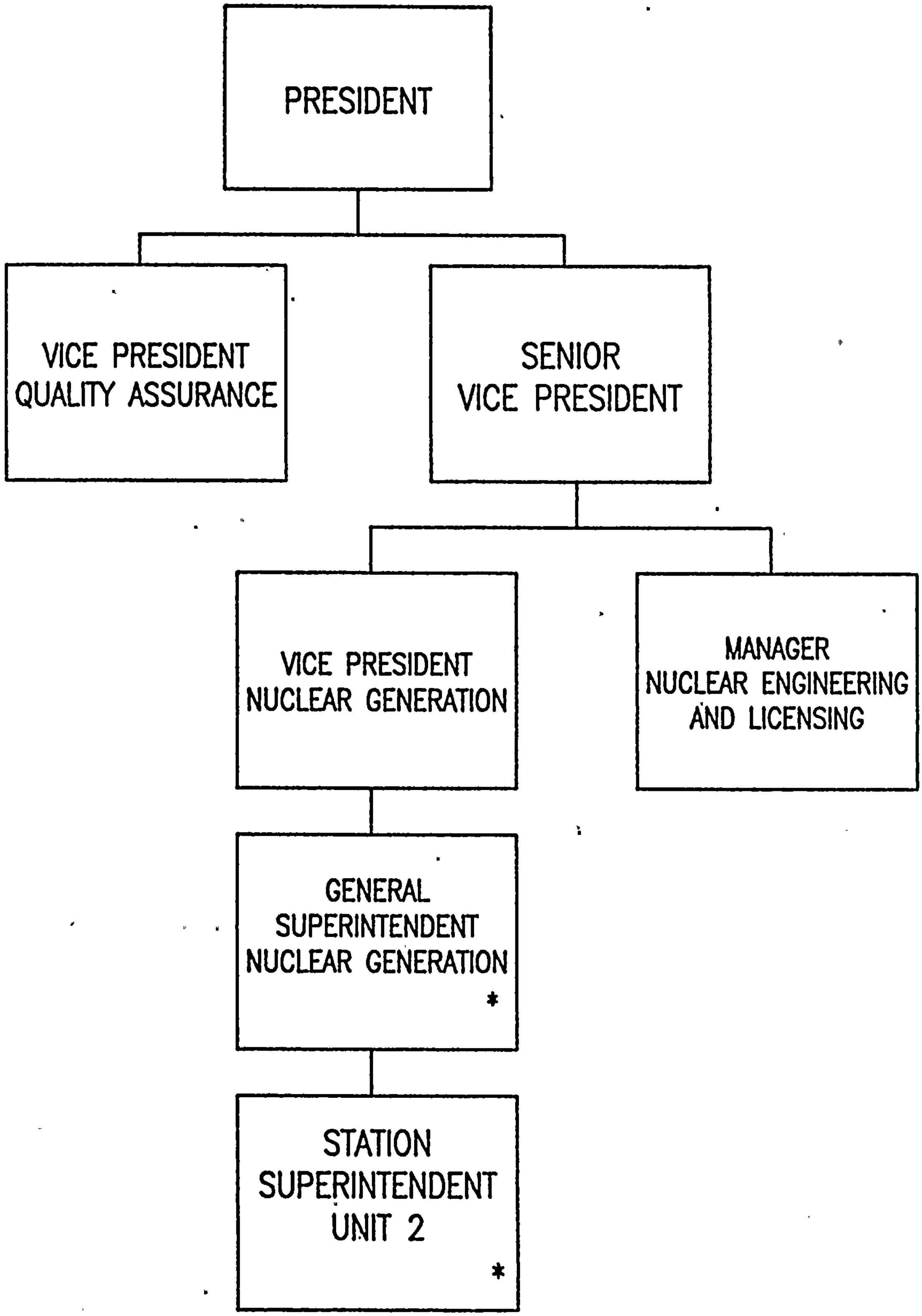
COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years of professional level experience in his/her field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The principal function of the ISEG is to examine plant operating characteristics and the various NRC and industry licensing and service advisories, and to recommend areas for improving plant operations or safety. The ISEG will perform independent review of plant activities, including maintenance, modifications, operational concerns, and analysis and make recommendations to the Supervisor Technical Support - Nuclear.

FIGURE 6.2.1-1 NIAGARA MOHAWK MANAGEMENT ORGANIZATION CHART



* Onsite positions.

Figure 6.2.2-1
NINE MILE POINT NUCLEAR SITE ORGANIZATION

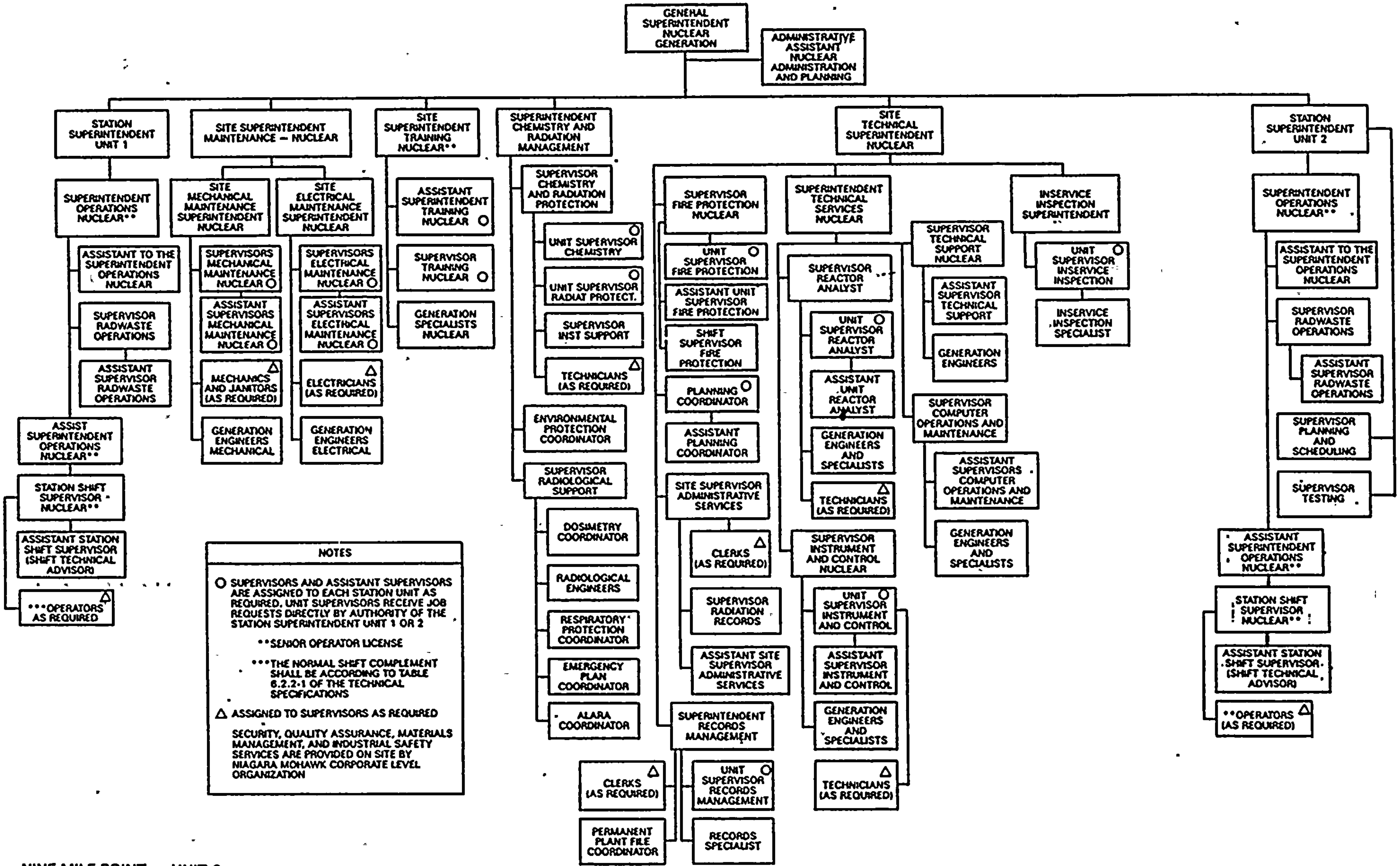


TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION(a)(b)

LICENSE	OPERATIONAL CONDITIONS 1	3, 4, 5	1, 2, 3, 4, 5	2
Senior Operator (Station Shift Supervisor)	1(d)	1(e)(d)	1(c)	1(d)
Operator	2	1(i),2(h)	2(c)	3
Unlicensed(f)	2	1(i),2(h)	3(c)	2
Senior Operator(g) (Asst. Station Shift Supervisor, Shift Technical Advisor Function)	1	1(h)	1(c)	1

TABLE NOTATIONS

(a) At any one time, more licensed or unlicensed operating people could be present for maintenance, repairs, refuel outages, etc.

(b) 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 because to an oncoming shift crewman scheduled to come on duty is late or absent.

(c) For operation longer than 8 hours without process computer.

(d) Any time the Shift Supervisor is absent from the control room while the unit is in OPERATIONAL CONDITION 1, 2, or 3, the Assistant Station Shift Supervisor when not in the STA function, or other individuals 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.

(e) An additional Senior Reactor Operator who has no other concurrent responsibilities shall supervise all core alterations.

(f) Those operating personnel not holding an Operator or Senior Operator license.

(g) The Assistant Station Shift Supervisor shall hold a Senior Operator's license and performs the Shift Technical Advisor function when the Site Emergency Plan is activated in OPERATIONAL CONDITIONS 1, 2, or 3.

(h) OPERATIONAL CONDITION 3 only.

(i) OPERATIONAL CONDITIONS 4 and 5 only.

ORGANIZATION

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager - Nuclear Technology.

6.2.4 ASSISTANT STATION SHIFT SUPERVISOR/SHIFT TECHNICAL ADVISOR

Normally the Assistant Station Shift Supervisor (ASSS) shall function in a dual role (SRO/STA) and assume the duties of the Shift Technical Advisor (STA) when the Emergency Plan is activated in OPERATIONAL CONDITIONS 1, 2 or 3. The STA 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 ASSS, when fulfilling the role of the STA, shall have a bachelor's degree in a physical science, engineering, or a PE license issued by examination, 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 FACILITY STAFF QUALIFICATIONS

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 Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The Licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980, NRC letter to all licensees.

6.4 TRAINING

A retraining and replacement training program for the unit staff shall be maintained under the direction of the Superintendent - Training Nuclear, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1978 and Appendix A of 10 CFR 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. A training program for the Fire Brigade shall be maintained under the direction of the Superintendent - Training Nuclear and the Supervisor - Fire Protection Nuclear and shall meet or exceed the requirements of Appendix R to 10 CFR 50.

6.5 REVIEW AND AUDIT

6.5.1 SITE OPERATIONS REVIEW COMMITTEE

FUNCTION

6.5.1.1 The Site Operations Review Committee (SORC) shall function to advise the General Superintendent - Nuclear Generation on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The SORC shall be composed of the

Chairman: General Superintendent - Nuclear Generation
Member: Station Superintendent - Nuclear Generation
Member: Technical Superintendent - Nuclear Generation
Member: Superintendent Technical Services - Nuclear
Member: Site Superintendent Maintenance - Nuclear
Member: Supervisor Instrument and Control - Nuclear
Member: Superintendent Chemistry and Radiation Management
Member: Supervisor Reactor Analysis
Member: Supervisor Technical Support
Member: Engineer

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SORC shall meet at least once every calendar month and as convened by the SORC Chairman or a designated alternate.

QUORUM

6.5.1.5 The quorum of the SORC necessary for the performance of the SORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or a designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The SORC shall be responsible for:

a. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Vice President - Nuclear Generation and to the Safety Review and Audit Board;

6.5 REVIEW AND AUDIT

6.5.1 SITE OPERATIONS REVIEW COMMITTEE

6.5.1.6 (Continued)

b. Review of all REPORTABLE EVENTS;

c. Review of unit operations to detect potential hazards to nuclear safety;

d. Performance of special reviews, investigations, or analyses and reports thereon as requested by the General Superintendent - Nuclear Generation or the Safety Review and Audit Board;

e. Safety evaluations and analyses resulting from technical review and control activities 6.5.2.1, 6.5.2.2, 6.5.2.3, and 6.5.2.5.

DUTIES

6.5.1.7 The SORC shall:

a. 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.

b. Provide written notification within 24 hours to the Vice President - Nuclear Generation and the Safety Review and Audit Board of disagreement between the SORC and the General Superintendent - Nuclear Generation; however, the General Superintendent - Nuclear Generation shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The SORC shall maintain written minutes of each SORC meeting that at a minimum, document the result of all SORC activities performed under the responsibilities and authority provisions of these Technical Specifications. Copies shall be provided to the Vice President - Nuclear Generation and the Safety Review and Audit Board.

6.5.2 TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.2.1 Each procedure and program required by Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group that prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group that prepared the procedure, or changes thereto. Approval of procedures and programs and changes thereto and their safety evaluations, shall be controlled by administrative procedures.

6.5.2.2 Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group that prepared the proposed change, but who may be from the same organization as the individual/group that prepared the proposed change.

REVIEW AND AUDIT

TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.2.2 (Continued)

Proposed changes to the Technical Specifications shall be approved by the General Superintendent - Nuclear Generation.

6.5.2.3 Proposed modifications to unit structures, systems, and components that affect nuclear safety shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group that designed the modification, but who may be from the same organization as the individual/group that designed the modification. Proposed modifications to structures, systems, and components and the safety evaluations shall be approved before implementation by the General Superintendent - Nuclear Generation; or the Station Superintendent - Nuclear Generation, or the Technical Superintendent - Nuclear Generation, as previously designated by the General Superintendent - Nuclear Generation.

6.5.2.4 Individuals responsible for reviews performed in accordance with Specifications 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the General Superintendent - Nuclear Generation to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary such review shall be performed by the appropriate designated station review personnel.

6.5.2.5 Proposed tests and experiments that affect station nuclear safety and are not addressed in the FSAR or Technical Specifications and their safety evaluations shall be reviewed by the General Superintendent - Nuclear Generation; or or by the Station Superintendent - Nuclear Generation, or the Technical Superintendent - Nuclear Generation, as previously designated by the General Superintendent - Nuclear Generation.

6.5.2.6 The General Superintendent - Nuclear Generation shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President - Nuclear Generation.

6.5.2.7 The facility security program and implementing procedures shall be reviewed at least every 12 months. Recommended changes shall be approved by the General Superintendent - Nuclear Generation and transmitted to the Vice President - Nuclear Generation, and to the Chairman of the Safety Review and Audit Board.

6.5.2.8 The facility emergency plan and implementing procedures shall be reviewed at least every 12 months. Recommended changes shall be approved by the General Superintendent - Nuclear Generation and transmitted to the Vice President - Nuclear Generation and to the Chairman of the Safety Review and Audit Board.

REVIEW AND AUDIT

TECHNICAL REVIEW AND CONTROL ACTIVITIES

6.5.2.9 The General Superindendent - Nuclear Generation shall assure the performance of a review by a qualified individual/organization of changes to the Radiological Waste Treatment systems.

6.5.2.10 Review of any accidental, unplanned, or uncontrolled radioactive release, including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

6.5.2.11 Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL. Approval of any changes shall be made by the General Superintendent - Nuclear Generation or a designee before implementation of such changes.

6.5.2.12 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained. Copies shall be provided to the Vice President - Nuclear Generation and the Safety Review and Audit Board.

6.5.3 SAFETY REVIEW AND AUDIT BOARD

FUNCTION

6.5.3.1 The Safety Review and Audit Board (SRAB) 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 and
i. Other appropriate fields associated with the unique characteristics of the nuclear power plant

The SRAB shall report to and advise the Vice President-Nuclear Generation and Manager-Nuclear Engineering and Licensing on those areas of responsibility in Specifications 6.5.3.7 and 6.5.3.8.

COMPOSITION

6.5.3.2 The SRAB shall be composed of the:

Chairman: Vice President, Manager or Staff Engineer
Member: General Superintendent - Nuclear Generation
Member: Staff Engineer - Nuclear
Member: Staff Engineer - Mechanical or Electrical
Member: Staff Engineer - Environmental
Member: Consultant (Specification 6.5.3.4)

REVIEW AND AUDIT

SAFETY REVIEW AND AUDIT BOARD

ALTERNATES

6.5.3.3 All alternate members shall be appointed in writing by the SRAB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRAB activities at any one time.

CONSULTANTS

6.5.3.4 Consultants shall be utilized as determined by the SRAB Chairman to provide expert advice to the SRAB.

MEETING FREQUENCY

6.5.3.5 The SRAB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.3.6 The quorum of the SRAB necessary for the performance of the SRAB review and audit functions of these Technical Specifications shall consist of the Chairman or the Chairman's designated alternate and at least three SRAB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.3.7 The SRAB shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

REVIEW AND AUDIT

SAFETY REVIEW AND AUDIT BOARD

REVIEW

6.5.3.7 (Continued)

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 SORC.

AUDITS

6.5.3.8 Audits of unit activities shall be performed under the cognizance of the SRAB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once every 12 months;

b. The performance, training, and qualifications of the entire unit staff at least once every 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 every 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once every 24 months;

e. The Unit Emergency Plan and implementing procedures at least once every 12 months.

f. The Unit Security Plan and implementing procedures at least once every 12 months.

g. The Radiological Environmental Monitoring Program and the results thereof at least once every 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once every 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once every 24 months;

REVIEW AND AUDIT

SAFETY REVIEW AND AUDIT BOARD

AUDITS

6.5.3.8 (Continued)

j. Any other area of unit operation considered appropriate by the SRAB or the Vice President - Nuclear Generation or the Manager - Nuclear Engineering and Licensing.

k. The Fire Protection Program and implementing procedures at least once per 24 months.

l. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

m. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

AUTHORITY

6.5.3.9 The SRAB shall report to and advise the Vice President - Nuclear Generation and Manager - Nuclear Engineering and Licensing on those areas of responsibility specified in Sections 6.5.3.7 and 6.5.3.8.

RECORDS

6.5.3.10 Records of SRAB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SRAB meeting shall be prepared, approved, and forwarded to the Vice President - Nuclear Generation and Manager - Nuclear Engineering and Licensing within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.3.7b, e, g, h shall be prepared, approved, and forwarded to the Vice President - Nuclear Generation and Manager - Nuclear Engineering and Licensing within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.3.8 shall be forwarded to the Vice President - Nuclear Generation, the Manager - Nuclear Engineering and Licensing, and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

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 50.72 and 10 CFR 50.73, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC, and the results of this review shall be submitted to the SRAB and the Vice President - Nuclear Generation.

6.7 SAFETY LIMIT VIOLATION

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 Vice President - Nuclear Generation and the SRAB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared in accordance with 10 CFR 50.73. The report shall be reviewed by the SORC. 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 within 30 days of the violation, and to the SRAB, and the Vice President - Nuclear Generation within 14 days.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978

b. The applicable procedures required to implement the requirements of NUREG-0737

c. Refueling operations

6.8 PROCEDURES AND PROGRAMS

PROCEDURES

6.8.1 (Continued)

d. Surveillance and test activities of safety-related equipment

e. Security Plan implementation

f. Emergency Plan implementation

g. Fire Protection Program implementation

h. PROCESS CONTROL PROGRAM implementation

i. OFFSITE DOSE CALCULATION MANUAL implementation and

j. Quality Assurance for effluent and environmental monitoring.

REVIEW AND APPROVAL

6.8.2 Each procedure of Specification 6.8.1, and changes thereto, shall be approved by the General Superintendent - Nuclear Generation based on the recommendations of SORC* before implementation and reviewed periodically as set forth in administrative procedures.

TEMPORARY CHANGES

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, approved by the General Superintendent - Nuclear Generation based on the recommendations of the SORC* within 14 days of implementation.

* SORC recommendations to the General Superintendent - Nuclear Generation are to be based on SORC responsibilities as identified in Specification 6.5.1.7.a.

PROCEDURES AND PROGRAMS

PROGRAMS

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, hydrogen recombiner, process sampling, containment and standby gas treatment systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel
2. Procedures for monitoring and
3. Provisions for maintenance of sampling and analysis equipment

c. Postaccident 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. This program shall include the following:

1. Training of personnel
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

d. Fire Protection Program

The Fire Protection Program is a program to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report as amended and as approved in the Safety Evaluation Report (NUREG-1047) dated February 1985 as supplemented. The noncompliances with the above Fire Protection Program that affect the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with the requirements of 10 CFR 50.73.

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 NRC Regional Office, 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 Subsection 14.2.12.2 and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted before March 1 of each year. The initial report shall be submitted before March 1 of the year after the plant achieves initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). The dose assignments to various duty functions may be estimated on the basis of pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totaling 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total

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* This tabulation supplements the requirements of 10 CFR 20.407.

REPORTING REQUIREMENTS

ROUTINE REPORTS

ANNUAL REPORTS

6.9.1.5 (Continued)

whole-body dose received from external sources should be assigned to specific major work functions.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours before the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed before exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Cleanup system flow history starting 48 hours before the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

c. Documentation of all challenges to safety/relief valves; and

d. Any other unit unique reports required on an annual basis.

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 monthly to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The initial report shall be submitted before May 1 of the year after the plant achieves initial criticality.

The Annual Radiological Environmental Operating Report 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,

* A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to all units at the site.

REPORTING REQUIREMENTS

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORTS

6.9.1.7 (Continued)

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 the land use census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table 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 Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplemental report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the Sampling Schedule of Table 3.12.1-1; and discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date the plant achieves initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power

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* 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 site. The submittal should combine those sections that are common to all units at the site; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

REPORTING REQUIREMENTS

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 (Continued)

Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR 61), type of container (e.g., LSA, Type A, Type B, Large Quantity), and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses from the radioactive liquid and gaseous effluents released from the unit 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 from their activities inside the SITE BOUNDARY (Figure 5.1.3-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 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 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the ODCM.

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.

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* In lieu of submission with the 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

6.9.1.8 (Continued)

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to liquid, gaseous, or solid radwaste treatment systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively, and a description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

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, of the Code of Federal Regulations (10 CFR), the following records shall be retained for at least the minimum period indicated.

6.10.1.1 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 submitted to the Commission

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications

e. Records of changes made to the procedures required by Specification 6.8.1

f. Records of radioactive shipments.

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

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

RECORD RETENTION

6.10.1.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories

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

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, and not listed in Specification 6.10.1.1

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59

k. Records of meetings of the SORC and the SRAB

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

m. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed

n. Records of unit radiation and contamination surveys

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2), each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them

c. An individual qualified in radiation protection (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 Radiation Protection Supervisor or the Radiation Protection Supervisor's designee 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 keyed access shall be maintained under the administrative control of the Station Shift Supervisor or the designee on duty and/or the Radiation Protection Supervisor or designee. 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 such radiation levels 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 drywell, where no enclosure exists that can be locked, and no enclosure can be reasonably constructed around the individual

* Measurements made at 18 inches from the source of radioactivity.

** Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA

6.12.2 (Continued)

areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM

6.13.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission before implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   3. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL

6.14.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission before implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed; each page should be numbered, dated, and marked with the revision number; appropriate analyses or evaluations justifying the change(s) should be included;

OFFSITE DOSE CALCULATION MANUAL

6.14.2 (Continued)

2. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3. Documentation of the fact that the SORC has reviewed the change and found it acceptable.

b. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radwaste treatment systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the SORC. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period that precedes the time when the change is to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

* Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS

6.15.1.a (Continued)

8. Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

APPENDIX B

TO FACILITY OPERATING LICENSE NO. NPF-54

NINE MILE POINT NUCLEAR STATION UNIT 2

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

NINE MILE POINT NUCLEAR STATION
UNIT NO. 2

ENVIRONMENTAL PROTECTION PLAN
(NONRADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of nonradiological environmental values during operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating Licensing Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES-OL which relate to water quality matters are regulated by way of the licensee's SPDES permit.

2.0 Environmental Protection Issues

In the FES-OL dated May 1985, the staff considered the environmental impacts associated with the operation of the Nine Mile Point Nuclear Station Unit No. 2. No aquatic/water quality, terrestrial, or noise issues were identified.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in station design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in station design or operation or performance of tests or experiments which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measurable nonradiological environmental effects are confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

* This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluated in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

The licensee shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of the Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

3.2 Reporting Related to the SPDES Permit and State Certification

Changes to, or renewals of, the SPDES Permit or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The licensee shall notify the NRC of changes to the effective SPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the SPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; fish kills; increase in nuisance organisms or conditions; unanticipated or emergency discharge of waste water or chemical substances, and damage to vegetation resulting from cooling tower drift deposition.

No routine monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the State of New York under the authority of the Clean Water Act for any requirements for aquatic monitoring.

4.2.2 Terrestrial Monitoring

No terrestrial monitoring is required.

4.2.3 Noise Monitoring

No noise monitoring is required.

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The period of the first report shall begin with the date of issuance of the operating license, and the initial report shall be submitted prior to May 1 of the year following issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 (if any) of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics; (b) describe the probable cause of the event; (c) indicate the action taken to correct the reported event; (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems; and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided with a copy of such report at the same time it is submitted to the other agency.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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