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Section 6 **ADMINISTRATIVE CONTROLS**

ADMINISTRATIVE CONTROLS

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1.0 DEFINITIONS

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

ACTION

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

AVERAGE PLANAR EXPOSURE (APE)

The AVERAGE PLANAR EXPOSURE (APE) 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 (APLHGR)

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 RATE(s) 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

A CHANNEL shall be an arrangement of a sensor and associated components used to evaluate plant variables and generate a single protective action signal. A CHANNEL terminates and loses its identity where single action signals are combined in a TRIP SYSTEM or logic system.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the CHANNEL output such that it responds with the necessary range and accuracy to known values of the parameter which the CHANNEL monitors. The CHANNEL CALIBRATION shall encompass the entire CHANNEL including the required 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

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 CHANNEL(s) measuring the same parameter.

1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

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

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated control cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

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

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/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."

1.0 DEFINITIONS

FRACTION OF RATED THERMAL POWER (FRTP)

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

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

FUEL DESIGN LIMITING RATIO (FDLRX)

The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum.

FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)

The FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) shall be the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary 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.

LIMITING CONTROL ROD PATTERN (LCRP)

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

LINEAR HEAT GENERATION RATE (LHGR)

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 (LSFT)

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps so that the entire logic system is tested.

1.0 DEFINITIONS

MINIMUM CRITICAL POWER RATIO (MCPR)

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting 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. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency 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 specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

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 UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

1.0 DEFINITIONS

PRIMARY CONTAINMENT INTEGRITY (PCI)

PRIMARY CONTAINMENT INTEGRITY (PCI) shall exist when:

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

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

2) Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

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

c. Each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

d. The primary containment leakage rates are maintained within the limits of Specification 3.7.A.

e. The suppression chamber is in compliance with the requirements of Specification 3.7.K.

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

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER (RTP) shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWT.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME shall be the time interval for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator.

1.0 DEFINITIONS

REPORTABLE EVENT

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

SECONDARY CONTAINMENT INTEGRITY (SCI)

SECONDARY CONTAINMENT INTEGRITY (SCI) shall exist when:

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

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

   2) Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as permitted by Specification 3.7.O.

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

c. The standby gas treatment system is in compliance with the requirements of Specification 3.7.P.

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

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

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.7.N.1.

SHUTDOWN MARGIN (SDM)

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

SOURCE CHECK

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

STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR)

The STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) shall be the limit which protects against exceeding the fuel end-of-life steady state design criteria.

1.0 DEFINITIONS

THERMAL POWER

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

TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR)

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be the limit which protects against fuel centerline melting and 1% plastic cladding strain during transient conditions throughout the life of the fuel.

TRIP SYSTEM

A TRIP SYSTEM shall be an arrangement of instrument CHANNEL trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A TRIP SYSTEM may require one or more instrument CHANNEL trip signals related to one or more plant parameters in order to initiate TRIP SYSTEM action. Initiation of protective action may require the tripping of a single TRIP SYSTEM or the coincident tripping of two TRIP SYSTEMs.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage in the primary containment which is not IDENTIFIED LEAKAGE.

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	NA	Not applicable

TABLE 1-2

OPERATIONAL MODES

MODE	MODE SWITCH POSITION[f]	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown[a,e]	> 212°F
4. COLD SHUTDOWN	Shutdown[a,b,e]	≤ 212°F
5. REFUELING[c]	Shutdown or Refuel[a,d]	≤ 140°F

TABLE NOTATIONS

(a) The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified individual.

(b) 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 3.10.I.

(c) Fuel in the reactor vessel with one or more vessel head closure bolts less than fully tensioned or with the head removed.

(d) See Special Test Exceptions 3.12.A and 3.12.B.

(e) The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided the one-rod-out interlock is OPERABLE.

(f) When there is no fuel in the reactor vessel, the reactor is considered not to be in any OPERATIONAL MODE. The reactor mode switch may then be in any position or may be inoperable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A 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 MODE(s) 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.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.7.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.7.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

Reactor Protection System (RPS) Instrumentation Setpoints

2.2.A The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.A-1.

APPLICABILITY: As shown in Table 3.1.A-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Trip Setpoint column of Table 2.2.A-1, declare the CHANNEL inoperable and apply the applicable ACTION statement requirement of Specification 3.1.A until the CHANNEL is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
1. Intermediate Range Monitor:	
a. Neutron Flux - High	≤120/125 divisions of full scale
b. Inoperative	NA
2. Average Power Range Monitor:	
a. Setdown Neutron Flux - High	≤15% of RATED THERMAL POWER
b. Flow Biased Neutron Flux - High	
1) Dual Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 62%, with a maximum of
b) High Flow Maximum	≤120% of RATED THERMAL POWER
2) Single Recirculation Loop Operation	
a) Flow Biased	≤0.58W[a] + 58.5%, with a maximum of
b) High Flow Maximum	≤116.5% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤120% of RATED THERMAL POWER
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤1060 psig
4. Reactor Vessel Water Level - Low	≥144 inches above top of active fuel
5. Main Steam Line Isolation Valve - Closure	≤10% closed
6. Main Steam Line Radiation - High	≤3[b] x normal full power background (without hydrogen addition)

a W shall be the recirculation loop flow expressed as a percentage of the recirculation loop flow which produces a rated core flow of 98 million lbs/hr.

b With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the primary coolant, this Unit 2 setting may be increased to "≤3 x full power background (with hydrogen addition)."

TABLE 2.2.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

Functional Unit	Trip Setpoint
7. Drywell Pressure - High	≤2 psig
8. Scram Discharge Volume Water Level - High	≤40.4 gallons (Unit 2) ≤41 gallons (Unit 3)
9. Turbine Stop Valve - Closure	≤10% closed
10. Turbine EHC Control Oil Pressure - Low	≥900 psig
11. Turbine Control Valve Fast Closure	≥460 psig EHC fluid pressure
12. Turbine Condenser Vacuum - Low	≥21 inches Hg vacuum
13. Reactor Mode Switch Shutdown Position	NA
14. Manual Scram	NA

BASES

2.1 SAFETY LIMITS

The Specifications in Section 2.1 establish operating parameters to assure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). These parameters are based on the Safety Limits requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

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 limit is set such that no calculated fuel damage would occur as a result of an AOO. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit for the MINIMUM CRITICAL POWER RATIO (MCPR) that represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforations 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 integrity Safety Limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the fuel cladding integrity Safety Limit is established such that no calculated fuel damage shall result from an abnormal operational transient. This is accomplished by selecting a MCPR fuel cladding integrity Safety Limit which assures that during normal operation and AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Exceeding a Safety Limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

BASES

2.1.A THERMAL POWER, Low Pressure or Low Flow

This fuel cladding integrity Safety Limit is established by establishing a limiting condition on core THERMAL POWER developed in the following method. At pressures below 800 psia (~785 psig), the core elevation pressure drop (0% power, 0% flow) is greater than 4.56 psi. At low powers and flows, this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28 x 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 4.56 psi driving head will be greater than 28 x 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. At 25% of RATED THERMAL POWER, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig is conservative.

2.1.B THERMAL POWER, High Pressure and High Flow

This fuel cladding integrity Safety Limit is set such that no (mechanistic) fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power ratio (CPR) 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 margin between a MCPR of 1.0 (onset of transition boiling) and the Safety Limit, is derived from a detailed statistical analysis which considers the uncertainties in monitoring the core operating state, including uncertainty in the critical power correlation. Because the transition boiling correlation is based on a significant quantity of practical test data, there is a very high confidence that operation of a fuel assembly at the condition where MCPR is equal to the fuel cladding integrity Safety Limit would not produce transition boiling. In addition, during single recirculation loop operation, the MCPR Safety Limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

However, if transition boiling were to occur, cladding perforation would not necessarily be expected. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicates that BWR fuel can survive for an extended period in an environment of transition boiling.

2.1.C Reactor Coolant System Pressure

The Safety Limit for the reactor coolant system pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The reactor coolant system pressure Safety Limit of 1345 psig, as measured by the vessel steam space pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressure vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure Safety Limit was chosen as the lower of the pressure transients permitted by the applicable design codes, ASME Boiler and Pressure Vessel Code Section III for the pressure vessel, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over design pressure (120% x 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirculation suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology to assure that this Safety Limit pressure is not exceeded for any reload is documented by the specific fuel vendor. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the primary system piping and provides similar margin of protection at the established pressure Safety Limit.

The normal operating pressure of the reactor coolant system is nominally 1000 psig. Both pressure relief and safety relief valves have been installed to keep the reactor vessel peak pressure below 1375 psig. However no credit is taken for relief valves during the postulated full closure of all MSIVs without a direct (valve position switch) scram. Credit, however, is taken for the neutron flux scram. The indirect flux scram and safety valve actuation provide adequate margin below the allowable peak vessel pressure of 1375 psig.

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2.1.D Reactor Vessel Water Level

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor 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 core cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds of the core height. The Safety Limit has been established at 12 inches above the top of the active irradiated fuel to provide a point which can be monitored and also provide adequate margin for effective action. The top of active fuel is 360 inches above vessel zero.

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2.2 LIMITING SAFETY SYSTEM SETTINGS

The Specifications in Section 2.2 establish operational settings for the reactor protection system instrumentation which initiates the automatic protective action at a level such that the Safety Limits will not be exceeded. These settings are based on the Limiting Safety System Settings requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(1):

> "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. "

2.2.A Reactor Protection System Instrumentation Setpoints

The Reactor Protection System (RPS) instrumentation setpoints specified in the table are the values at which the reactor scrams are set for each parameter. The scram settings 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 assist in mitigating the consequences of accidents. Conservatism incorporated into the transient analysis is documented by each approved fuel vendor. The bases for individual scram settings are discussed in the following paragraphs.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of eight chambers, four in each of the reactor protection system logic CHANNELs. The IRM is a 5 decade, 10 range, instrument which covers the range of power level between that covered by the SRM and the APRM. The IRM scram setting at 120 of 125 divisions is active in each range of the IRM. For example, if the instrument were on Range 1, the scram setting would be 120 divisions for that range; likewise, if the instrument were on Range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provides adequate protection against the single rod withdrawal error, a range of rod withdrawal events has been analyzed. This analysis included starting the event at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM CHANNEL closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power is limited to 1% of rated power, thus maintaining MCPR above the fuel cladding integrity Safety Limit. Based on the above analysis, the IRM provides protection against local

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control rod withdrawal errors and continuous withdrawal of control rods in the sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during Startup, a reduced power level, i.e., setdown, APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setting and the Safety Limit. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor; cold water from sources available during startup are not much colder than that already in the system; temperature coefficients are small; and, control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, 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 scram setting, the rate of power rise is no more than 5% of RATED THERMAL POWER per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the Safety Limit. The 15% APRM setdown scram setting remains active until the mode switch is placed in the Run position.

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, also provides a flow biased neutron flux which reads in percent of RATED THERMAL POWER. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. During abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram setting for dual recirculation loop operation, or with a 116.5% scram setting for single recirculation loop operation, none of the abnormal operational transients analyzed violates the fuel cladding integrity Safety Limit, and there is a substantial margin from fuel damage. One of the neutron flux scrams is flow dependent until it reaches the applicable setting where it is "clamped" at its maximum allowed value. The use of the flow referenced neutron flux scram setting provides additional margin beyond the use of a the fixed high flux scram setting alone.

An increase in the APRM scram setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit, yet allows operating margin that reduces the possibility of unnecessary scrams.

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During single recirculation loop operation, the normal drive flow relationship is altered as a result of reverse flow through the idle loop jet pumps when the active loop recirculation pump speed is above approximately 40% of rated. The core receives less flow than would be predicted based upon the dual recirculation loop drive flow to core flow relationship, and the APRM flow biased scram settings must be altered to continue to provide a reactor scram at a conservative neutron flux.

The scram setting must also be adjusted to ensure that the LHGR transient limit is not violated for any power distribution. The scram setting is adjusted in accordance with Specification 3/4.11.B in order to maintain adequate margin for the Safety Limit and yet allow operating margin sufficient to reduce the possibility of an unnecessary shutdown. The adjustment may also be accomplished by increasing the APRM gain. This provides the same degree of protection as reducing the scram settings by raising the initial APRM readings closer to the scram settings such that a scram would be received at the same point in a transient as if the scram settings had been reduced.

3. Reactor Vessel Steam Dome Pressure - High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The scram will quickly reduce the neutron flux, counteracting the pressure increase. The scram setting is slightly higher than the operating pressure to permit normal operation without spurious scrams. The scram setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement (reactor vessel steam space) compared to the highest pressure that occurs in the system during a transient.

In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus, exceeding the pressure Safety Limit. The pressure scram is available as backup protection to the high flux scram. Analyses are performed for each reload to assure that the pressure Safety Limit is not exceeded.

4. Reactor Vessel Water Level - Low

The reactor vessel water level scram setting was chosen far enough below the normal operating level to avoid spurious scrams but high enough above the fuel to assure that there is adequate protection for the fuel cladding integrity and reactor coolant system pressure Safety Limits. The scram setting is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

The scram setting provided is the actual water level which may be different than the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will

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decrease as power is increased to 100% in comparison to the level outside the shroud, to a maximum of seven inches, due to the pressure drop across the steam dryer. Therefore, at 100% power, an indicated water level of +8 inches water level may be as low as +1 inches inside the shroud which corresponds to 144 inches above the top of active fuel and 504 inches above vessel zero.

5. Main Steam Line Isolation Valve - Closure

Automatic isolation of the main steam lines is provided to give protection against rapid reactor depressurization and cooldown of the vessel. When the main steam line isolation valves begin to close, a scram signal provides for reactor shutdown so that high power operation at low reactor pressures does not occur. With the scram setting at 10% valve closure (from full open), there is no appreciable increase in neutron flux during normal or inadvertent isolation valve closure, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than the MSIV closure setting requires the reactor mode switch to be in the Startup/Hot Standby position, where protection of the fuel cladding integrity Safety Limit is provided by the IRM and APRM high neutron flux scram signals. Thus, the combination of main steam line low pressure isolation and the isolation valve closure scram with the mode switch in the Run position assures the availability of the neutron flux scram protection over the entire range of applicability of fuel cladding integrity Safety Limit.

6. Main Steam Line Radiation - High

High radiation levels in the main steam line tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. When high radiation is detected, a scram is initiated to mitigate the failure of fuel cladding. The scram setting is high enough above background radiation levels to prevent spurious scrams yet low enough to promptly detect gross failures in the fuel cladding. This setting is determined based on normal full power background (NFPB) radiation levels without hydrogen addition. With the injection of hydrogen into the feedwater for mitigation of intergranular stress corrosion cracking, the full power background levels may be significantly increased. The setting is increased based on the new background levels to allow for the injection of hydrogen. This trip function provides an anticipatory scram to limit offsite dose consequences, but is not assumed to occur in the analysis of any design basis event.

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7. Drywell Pressure - High

High pressure in the drywell could indicate a break in the primary pressure boundary systems or a loss of drywell cooling. Therefore, pressure sensing instrumentation is provided as a backup to the water level instrumentation. The reactor is scrammed on high pressure in order to minimize the possibility of fuel damage and reduce the amount of energy being added to the coolant and the primary containment. The scram setting was selected as low as possible without causing spurious scrams.

8. Scram Discharge Volume Water Level - High

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this system is an individual instrument volume for each of the scram discharge volumes. These two instrument volumes and their piping can hold in excess of 90 gallons of water and are the low point in the piping. No credit was taken for the instrument volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the scram discharge volumes are empty; however, should either scram discharge volume accumulate water, the water discharged to the piping from the reactor during a scram may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both instrument volumes which will alarm and scram the reactor while sufficient volume remains to accommodate the discharged water. Diverse level sensing methods have been incorporated into the design and logic of the system to prevent common mode failure. The setting for this anticipatory scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram, even with 5 gpm leakage per drive into the scram discharge volume. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the amount of insertion of the control rods.

9. Turbine Stop Valve - Closure

The turbine stop valve closure scram setting anticipates the pressure, neutron flux, and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity Safety Limit, even during the worst-case transient that assumes the turbine bypass fails to operate.

10. Turbine EHC Control Oil Pressure - Low

The turbine EHC control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the turbine control valve fast closure scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and reactor high pressure scrams. However, to provide the same margins as provided for the generator load rejection on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This scram anticipates the pressure transient which would be caused by imminent control valve closure and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the turbine control valve fast closure scram. However, since the control valves will not start to close until the fluid pressure is approximately 600 psig, the scram on low turbine EHC control oil pressure occurs well before turbine control valve closure begins. The scram setting is high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams.

11. Turbine Control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass valves; i.e., MCPR remains above the fuel cladding integrity Safety Limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides a wide margin to the value corresponding to 1% plastic strain of the cladding.

The scram setting based on EHC fluid pressure was developed to ensure that the pressure switch is actuated prior to the closure of the turbine control valves (at approximately 400 psig EHC fluid pressure), yet assure that the system is not actuated unnecessarily due to EHC system pressure transients which may cause EHC system pressure to momentarily decrease.

12. Turbine Condenser Vacuum - Low

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise and an increase in surface heat flux. To prevent the fuel cladding integrity Safety Limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the fuel cladding integrity Safety Limit from being exceeded, in the event of a turbine trip transient with bypass closure. The condenser low vacuum scram is anticipatory to the stop valve closure scram and causes a scram before the stop valves (and bypass valves) are closed and thus, the resulting transient is less severe.

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13. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant CHANNEL to the automatic protective instrumentation CHANNEL(s) and provides additional manual reactor scram capability.

14. Manual Scram

The manual scram is a redundant CHANNEL to the automatic protective instrumentation CHANNEL(s) and provides manual reactor scram capability.

3.0 - LIMITING CONDITIONS FOR OPERATION

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

B. 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.

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

1. At least HOT SHUTDOWN within the next 12 hours, and
2. 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 MODE 4 or 5.

D. When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. Exceptions to these requirements are stated in the individual Specifications.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE 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. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

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

   1. 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 Part 50, Section 50.55a(g) and 50.55a(f), respectively, except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i) or 50.55a(f)(6)(i), respectively.

4.0 - SURVEILLANCE REQUIREMENTS

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

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

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

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

6. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

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Specifications 3.0.A through 3.0.D establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.A establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODE(s) or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these ACTION(s) are not completed within the allowable outage time limits, a shutdown is required to place the facility in a reactor OPERATIONAL MODE or other specified condition in which the specification no longer applies.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered an OPERATIONAL MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.B establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION

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requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirement within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition for Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.C establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown condition when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Condition for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODE(s) of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant transient that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required ACTION(s).

The time limits of Specification 3.0.C allow 37 hours for the plant to be in COLD SHUTDOWN when a shutdown is required during POWER OPERATION. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other OPERATIONAL MODE, is not reduced. For example, if HOT SHUTDOWN is reached in 10 hours, the time allowed to reach COLD SHUTDOWN is the next 27 hours because the total time to reach COLD SHUTDOWN is not reduced from the allowable limit of 37 hours. Therefore, if remedial measures are completed that would permit a return to POWER OPERATION, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into an OPERATIONAL MODE or condition of operation for another specification in which the

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requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.C do not apply in MODES 4 or 5, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.D establishes limitations on a change in OPERATIONAL MODE(s) when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply to comply with the ACTION requirements if a change in MODE(s) were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODE(s) of operation or other specified conditions are not entered when corrective ACTION is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a change in OPERATIONAL MODE(s). Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.D do not apply because they would delay placing the facility in a lower MODE of operation.

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Specifications 4.0.A through 4.0.E establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.A establishes the requirement that surveillances must be performed during the OPERATIONAL MODE(s) or other conditions for which the requirements of the Limiting Condition for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor OPERATIONAL MODE or other specified condition for which the individual Limiting Condition for Operations are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.B establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified with and 18 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. Likewise, it is not the intent that refueling outage surveillances be performed during power operation unless it is consistent with safe plant operation. The limitation of Specification 4.0.B is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.C establishes that the failure to satisfy a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, is a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.C. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.B, was a violation of the OPERABILITY requirements for a Limiting Condition for Operation that is subject to enforcement action. The failure to perform a surveillance within the provisions of Specification 4.0.B is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements (e.g., in Specification 3.0.C), a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown would be required to comply with ACTION requirements or before other remedial measures would be required that may preclude the completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.D is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.D establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into an OPERATIONAL MODE or other specified condition for which these systems and components ensure safe operation of the

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facility. This provision applies to changes in OPERATIONAL MODE(s) or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to assure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION statements, the provisions of Specification 4.0.D do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.E establishes the requirement 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 shall 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. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for 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 the 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. The requirements of Specification 4.0.D to perform surveillance requirements before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, which is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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3/4.1.A 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 primary system, and

c. minimize the energy which must be absorbed and prevent criticality following a loss-of-coolant accident.

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 CHANNEL(s) may be out-of-service because of maintenance. When necessary, one CHANNEL may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent TRIP SYSTEM(s), each having a minimum of two CHANNEL(s) of tripping devices. Each CHANNEL has an input from at least one instrument CHANNEL which monitors a critical parameter. The outputs of the CHANNEL(s) are combined in a one-out- of-two-logic, i.e., an input signal on either one or both of the CHANNEL(s) will cause a TRIP SYSTEM trip. The outputs of the TRIP SYSTEM(s) are arranged so that a trip on both systems is required to produce a reactor scram. This system meets the intent of the proposed IEEE 279, "Standard for Nuclear Power Plant Protection Systems" issued September 13, 1966. The system has a reliability greater than that of a two-out-of-three system and somewhat less than that of a one-out-of-two system (reference APED 5179). The bases for the trip settings of the RPS are discussed in the Bases for Specification 2.2.A.

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Sections 7.4.4.2 and 7.4.4.3). During refueling, the primary Neutron Monitoring System (NMS) indication of neutron flux levels is provided by the Source Range Monitors (SRM). The SRMs provide input to the RPS, but shorting links are installed across the normally closed contacts such that tripping an SRM CHANNEL does not result in the trip of the RPS CHANNEL. To activate the SRM scram function, these shorting links must be removed from the RPS. The SRM control rod scram provides backup protection to refueling interlocks and SHUTDOWN MARGIN should a prompt reactivity excursion occur. Although the IRM and APRM functions are required to be OPERABLE during refueling, the SRMs provide the only on-scale monitoring of neutron flux levels during refueling and therefore the shorting links must be removed to enable the scram function of the SRMs. The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems (1 out of n). However, 3 IRM CHANNELs and 2 APRM CHANNELs per TRIP SYSTEM are still required, i.e. a minimum of 6 IRMs and 4 APRMs.

The RPS (and therefore removal of the RPS shorting links) is required to be OPERABLE in Refuel only with any control rod withdrawn from a core cell containing one or more fuel assemblies. Control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity

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of the core and therefore are not required to have the capability to scram. If all control rods are inserted, the RPS function is not required. In this condition, the required SHUTDOWN MARGIN and the one-rod-out interlock provide assurance that the reactor will not become critical. If the SHUTDOWN MARGIN has been demonstrated, the RPS shorting links are not required to be removed. Under these conditions, the capability of the one-rod-out interlock to prevent criticality has been demonstrated and the scram protection provided by the IRMs is sufficient to ensure a highly reliable scram if required.

In the power range, the APRM system provides required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required (and is automatically bypassed) in OPERATIONAL MODE 1, the APRMs cover only the intermediate and power range; and the IRMs provide adequate coverage in the startup and intermediate range. The IRM inoperative function ensures that the instrument CHANNEL fails in the tripped condition upon loss of detector voltage.

Three APRM instrument CHANNEL(s) are provided for each TRIP SYSTEM. APRM CHANNEL(s) #1 and #3 operate contacts in one logic path and APRM CHANNEL(s) #2 and #3 operate contacts in the other logic path of the TRIP SYSTEM. APRM CHANNEL(s) #4, #5 and #6 are arranged similarly in the other TRIP SYSTEM's dual logic paths. Each TRIP SYSTEM has one more APRM than is necessary to meet the minimum number required per CHANNEL. This allows the bypassing of one APRM per TRIP SYSTEM for maintenance, testing, or calibration. Additional IRM CHANNEL(s) have also been provided to allow for bypassing of one such CHANNEL.
A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status (reference SAR Section 7.7.1.2). A bypass in the Refuel or Startup/Hot Standby operational modes is provided for the turbine condenser low vacuum scram and main steam line isolation valve closure scrams for flexibility during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in Startup/Hot Standby.

The manual scram function is available in OPERATIONAL MODE(s) 1 through 5, thus providing for a manual means of rapidly inserting control rods whenever fuel is in the reactor.

The turbine stop valve closure scram, the turbine EHC control oil low pressure scram, and the turbine control valve fast closure scram occur by design on turbine first stage pressure which is normally equivalent to ~45% RATED THERMAL POWER. However, since this is dependent on bypass valve position, the conservative reactor power is used to determine applicability.

Surveillance requirements for the reactor protection system are selected in order to demonstrate proper function and operability. The surveillance intervals are determined in many different ways, such as, 1) operating experience, 2) good engineering judgement, 3) reliability analyses, or 4) other analyses that are found acceptable to the NRC. The performance of the specified surveillances at the specified frequencies provides assurance that the protective functions associated with each CHANNEL can be completed as assumed in the safety analyses. A surveillance interval of "prior to startup" assures that these functions are available to perform their safety functions during control

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rod withdrawal, and hence these surveillances must be completed prior to initiating control rod withdrawal for the purpose of "approach to criticality".

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

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 MODE(s) and at the frequencies shown in Table 4.2.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

a 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 3.2.A-1 for that trip function shall be taken.

3.2 - LIMITING CONDITIONS FOR OPERATION

3. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEMS, place at least one TRIP SYSTEM[b] in the tripped condition[c] within one hour and take the ACTION required by Table 3.2.A-1.

4.2 - SURVEILLANCE REQUIREMENTS

b If more CHANNEL(s) are inoperable in one TRIP SYSTEM than in the other, select the TRIP SYSTEM with the greater number of inoperable CHANNEL(s) to place in the tripped condition except when this would cause the trip function to occur; if both TRIP SYSTEM(s) have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.

c 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 one hour or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit		Trip Setpoint[i]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1.	PRIMARY CONTAINMENT ISOLATION				
a.	Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b.	Drywell Pressure - High[d]	≤2 psig	2	1, 2, 3	20
c.	Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2.	SECONDARY CONTAINMENT ISOLATION				
a.	Reactor Vessel Water Level - Low[c]	≥144 inches	2	1, 2, 3 & *	24
b.	Drywell Pressure - High[c,d]	≤2 psig	2	1, 2, 3	24
c.	Reactor Building Ventilation Exhaust Radiation - High[c]	≤4 mR/hr	2	1, 2, 3 & **	24
d.	Refueling Floor Radiation - High[c]	≤100 mR/hr	2	1, 2, 3 & **	24
3.	MAIN STEAM LINE (MSL) ISOLATION				
a.	Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b.	MSL Tunnel Radiation - High[b]	≤3[g] x normal background	2	1, 2, 3	21
c.	MSL Pressure - Low	≥825 psig	2	1	22
d.	MSL Flow - High	≤120% of rated	2/line	1, 2, 3	21
e.	MSL Tunnel Temperature - High	≤200°F	2 of 4 in each of 2 sets	1, 2, 3	21

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[i]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation[f]	NA	NA	1, 2, 3	23
b. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	23
5. ISOLATION CONDENSER ISOLATION				
a. Steam Flow - High	≤300% of rated steam flow	1	1, 2, 3	23
b. Return Flow - High	≤32 (Unit 2)/ ≤14.8 (Unit 3) inches water diff.	1	1, 2, 3	23
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	≤300% of rated steam flow [h]	1	1, 2, 3	23
b. Reactor Vessel Pressure - Low	≥80 psig	2	1, 2, 3	23
c. Area Temperature - High	≤200°F	4[j]	1, 2, 3	23

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[i]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
7. SHUTDOWN COOLING ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	3, 4, 5	23
b. Recirculation Line Water Temperature - High (Cut-in Permissive)	≤350°F	2[a]	1, 2, 3	23

TABLE 3.2.A-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 8 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Be in at least STARTUP within 8 hours.

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

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

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains isolation actuation capability.

(b) Also trips the mechanical vacuum pump and isolates the steam jet air ejectors.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

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

(e) Only one TRIP SYSTEM.

(f) Closes only reactor water cleanup system isolation valves.

(g) Normal background is as measured during full power operation without hydrogen being injected. With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the feedwater, this Unit 2 setting may be as measured during full power operation with hydrogen being injected.

(h) Includes a time delay of $3 \leq t \leq 9$ seconds.

(i) Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(j) All four switches in either of 2 groups for each trip system.

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	M	Q	1, 2, 3
c. Drywell Radiation - High	S	M	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c]	S	M	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c]	NA	M	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c]	S	M	E	1, 2, 3 & **
d. Refueling Floor Radiation - High[c]	S	M	Q	1, 2, 3 & **
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	M	E[a]	1, 2, 3
b. MSL Tunnel Radiation - High	S	M	E[d]	1, 2, 3
c. MSL Pressure - Low	NA	M	Q	1
d. MSL Flow - High	S	M	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
5. ISOLATION CONDENSER				
a. Steam Flow - High	NA	M	Q	1, 2, 3
b. Return Flow - High	NA	M	Q	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	M	E[a]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	E[a]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. SHUTDOWN COOLING ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	3, 4, 5
b. Recirculation Line Water Temperature - High (Cut-in Permissive)	NA	M	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

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

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed every 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.
3. With either ADS TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within:
   a. 7 days provided that both the HPCI and IC are OPERABLE, or
   b. 72 hours.

   With the above provisions of this ACTION not met, be in at least HOT

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

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 MODE(s) and at the frequencies shown in Table 4.2.B-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.2 - SURVEILLANCE REQUIREMENTS

TABLE 3.2.B-1

EMERGENCY CORE COOLING SYSTEMS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low[b]	≥84 inches	4	1, 2, 3, 4[c], 5[c]	30
b. Drywell Pressure - High[b][f]	≤2 psig	4	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	≥300 psig & ≤350 psig	2	1, 2, 3 4[c], 5[c]	31 32
d. CS Pump Discharge Flow - Low (Bypass)	≥750 gpm	1/loop	1, 2, 3, 4[c], 5[c]	33
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	≥84 inches	4	1, 2, 3, 4[c], 5[c]	30
b. Drywell Pressure - High[f]	≤2 psig	4	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	≥300 psig & ≤350 psig	2	1, 2, 3 4[c], 5[c]	31 32
d. LPCI Pump Discharge Flow - Low (Bypass)	≥1000 gpm	1/loop	1, 2, 3, 4[c], 5[c]	33

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	4	1, 2, 3	35
b. Drywell Pressure - High[f]	≤2 psig	4	1, 2, 3	35
c. Condensate Storage Tank Level - Low[i]	≥10,000 gal	2	1, 2, 3	35
d. Suppression Chamber Water Level - High[i]	≤15′ 5" above bottom of chamber	2	1, 2, 3	35
e. Reactor Vessel Water Level - High (Trip)	≤194 inches	1	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	≥600 gpm	1	1, 2, 3	33
g. Manual Initiation	NA	1/system	1, 2, 3	34
4. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'A'[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	30
b. Drywell Pressure - High[f]	≤2 psig	2	1, 2, 3	30
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤10 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
5. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'B'[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	30
b. Drywell Pressure - High[f]	≤2 psig	2	1, 2, 3	30
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤10 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2930±146 volts decreasing voltage	2/bus	1, 2, 3, 4[e], 5[e]	36
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3784 volts (Unit 2)[g][j] ≥ 3832 volts (Unit 3)[g][j]	2/bus	1, 2, 3, 4[e], 5[e]	36

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the associated ECCS system(s) inoperable.

b. With more than one CHANNEL inoperable, declare the associated ECCS system(s) inoperable.

ACTION 31 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement:

a. For ADS, declare the associated ADS TRIP SYSTEM inoperable.

b. For CS, LPCI or HPCI, declare the associated ECCS system(s) inoperable.

ACTION 32 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour.

ACTION 33 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour; restore the inoperable CHANNEL to OPERABLE status within 7 days or declare the associated ECCS system(s) inoperable.

ACTION 34 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, restore the inoperable CHANNEL to OPERABLE status within 8 hours or declare the associated ECCS system(s) inoperable.

ACTION 35 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place at least one inoperable CHANNEL in the tripped condition within one hour or declare the HPCI system inoperable.

ACTION 36 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour, or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.9.A or 3.9.B, as appropriate.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

TABLE NOTATION

(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the associated Functional Unit maintains ECCS initiation capability.

(b) Also actuates the associated emergency diesel generator.

(c) When the system is required to be OPERABLE per Specification 3.5.B.

(d) Not required to be OPERABLE when reactor steam dome pressure is ≤ 150 psig.

(e) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

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

(g) With no LOCA signal present, there is an additional time delay of 5 ± 0.25 minutes.

(h) Reactor water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(i) Provides signal to pump suction valves only.

(j) There is an inherent time delay of 7 ± 1.4 seconds on degraded voltage.

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. CS Pump Discharge Flow - Low (Bypass)	NA	Q	Q[e]	1, 2, 3, 4[b], 5[b]
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	Q	Q[e]	1, 2, 3, 4[b], 5[b]
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	M	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	Q[f]	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	Q	Q	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. AUTOMATIC DEPRESSURIZATION SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q[f]	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Initiation Timer	NA	E	E	1, 2, 3
d. Low Low Level Timer	NA	E	E	1, 2, 3
e. CS Pump Discharge Pressure - High (Permissive)	NA	M	Q	1, 2, 3
f. LPCI Pump Discharge Pressure - High (Permissive)	NA	M	Q	1, 2, 3
5. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	NA	E	E	1, 2, 3, 4[c], 5[c]

TABLE 4.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

(b) When the system is required to be OPERABLE per Specification 3.5.B.

(c) Required when the associated diesel generator is required to be OPERABLE per Specification 3.9.B.

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

(e) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(f) Unit 2 transmitters are calibrated once per 18 months. Unit 2 trip units and Unit 3 level switches are calibrated at the frequency indentified in the table.

3.2 - LIMITING CONDITIONS FOR OPERATION

C. ATWS - RPT

The anticipated transient without scram recirculation pump trip (ATWS - RPT) instrumentation CHANNEL(s) shown in Table 3.2.C-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

1. With an ATWS - RPT instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.C-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one level CHANNEL or one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), within 14 days, either restore the inoperable CHANNEL to OPERABLE status or place the inoperable CHANNEL in the tripped condition[a]. Otherwise, be in STARTUP within the next 6 hours.
3. With two level CHANNELS or two pressure CHANNELS inoperable in one or both TRIP SYSTEM(s), declare the TRIP SYSTEM(s) inoperable.

4.2 - SURVEILLANCE REQUIREMENTS

C. ATWS - RPT

1. Each ATWS - RPT 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 4.2.C-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

---

a The inoperable CHANNEL(s) need not be placed in the tripped condition where this would cause the Trip Function to occur.

3.2 - LIMITING CONDITIONS FOR OPERATION

4. With one level CHANNEL and one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), restore at least one inoperable CHANNEL to OPERABLE status within 14 days or be in STARTUP within the next 6 hours.

5. 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.

6. With both TRIP SYSTEM(s) inoperable, restore at least one TRIP SYSTEM to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

4.2 - SURVEILLANCE REQUIREMENTS

TABLE 3.2.C-1

ATWS - RPT INSTRUMENTATION

Functional Unit	Trip Setpoint[c]	Minimum CHANNEL(s) per TRIP SYSTEM[a]
1. Reactor Vessel Water Level - Low Low	≥84 inches[b]	2
2. Reactor Vessel Pressure - High	≤1250 psig	2

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 Includes a time delay of $8 \le t \le 10$ seconds.

c Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

TABLE 4.2.C-1

ATWS - RPT INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Water Level - Low Low	S	Q	E[a]
2. Reactor Vessel Pressure - High	S	Q	E[a]

a Trip units are calibrated at least once per 92 days and transmitters are calibrated at the frequency identified in the table.

3.2 - LIMITING CONDITIONS FOR OPERATION

D. Isolation Condenser Actuation

The isolation condenser actuation instrumentation CHANNEL(s) shown in Table 3.2.D-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with the reactor steam dome pressure >150 psig.

ACTION:

1. With an isolation condenser actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.D-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more isolation condenser system actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.D-1.

4.2 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser Actuation

1. Each isolation condenser 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 4.2.D-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per TRIP SYSTEM[a]	ACTION
Reactor Vessel Pressure - High	$\leq$1070 psig	2	40

ACTION

ACTION 40 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the isolation condenser system inoperable.

b. With more than one CHANNEL inoperable, declare the isolation condenser system inoperable.

---

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.

TABLE 4.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
Reactor Vessel Pressure - High	NA	M	Q

3.2 - LIMITING CONDITIONS FOR OPERATION

E. Control Rod Block Actuation

The control rod block actuation instrumentation CHANNEL(s) shown in Table 3.2.E-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.E-1.

ACTION:

1. With a control rod block actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.E-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, take the ACTION required by Table 3.2.E-1.

4.2 - SURVEILLANCE REQUIREMENTS

E. Control Rod Block Actuation

Each of the required control rod block actuation TRIP SYSTEM(s) and instrumentation CHANNEL(s) shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.E-1.

TABLE 3.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
1. ROD BLOCK MONITORS[a]				
a. Upscale	As specified in the COLR	2	1[e]	50
b. Inoperative	NA	2	1[e]	50
c. Downscale	$\geq 5/125$ of full scale	2	1[e]	50
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	$\leq (0.58W+50)$[g]	4	1	51
2. Single Recirculation Loop Operation	$\leq (0.58W+46.5)$[g]	4	1	51
b. Inoperative	NA	4	1, 2, 5[h]	51
c. Downscale	$\geq 3/125$ of full scale	4	1	51
d. Startup Neutron Flux - High	$\leq 12/125$ of full scale	4	2, 5[h]	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	6	2, 5	51
b. Upscale	$\leq$108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	$\geq$5/125 of full scale	6	2, 5	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	(Unit 2) ≤29 gal (Unit 3) ≤25 gal	1 per bank	1, 2, 5[f]	52
b. SDV Switch in Bypass	NA	1	5[f]	52

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 50 - Declare the rod block monitor inoperable and take the ACTION required by Specification 3.3.M.

ACTION 51- With the number of OPERABLE CHANNEL(s):

a. One less than required by the Minimum CHANNEL(s) 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 CHANNEL(s) per Trip Function requirement, place at least one inoperable CHANNEL in the tripped condition within one hour.

ACTION 52 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per Trip Function requirement, place the inoperable CHANNEL in the tripped condition within one hour.

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

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

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

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

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

(e) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(f) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) The Average Power Range Monitor rod block function is varied as a function of recirculation drive flow (W). The trip setting of this function must be maintained in accordance with Specification 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98 x10^6 lbs/hr.

(h) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(i) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains control rod block capability.

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], M	Q	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2,[i] 5
b. Upscale[g]	NA	S/U[b], W	E	2,[i] 5
c. Inoperative[g]	NA	S/U[b], W	NA	2,[i] 5

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

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

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

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

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

3.2 - LIMITING CONDITIONS FOR OPERATION

F. Accident Monitoring

The accident monitoring instrumentation CHANNEL(s) shown in Table 3.2.F-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.2.F-1.

ACTION:

With one or more of the required number of accident monitoring instrumentation CHANNEL(s) inoperable, take the ACTION shown by Table 3.2.F-1.

4.2 - SURVEILLANCE REQUIREMENTS

F. Accident Monitoring

Each of the required accident monitoring instrumentation CHANNEL(s) shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.F-1.

TABLE 3.2.F-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	Required CHANNEL(s)	Minimum CHANNEL(s)	Applicable OPERATIONAL MODE(s)	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	60
2. Reactor Vessel Water Level	2	1	1, 2	60
3. Torus Water Level	2	1	1, 2	60
4. Torus Water Temperature	2	1	1, 2	60
5. Drywell Pressure - Wide Range	2	1	1, 2	60
6. Drywell Pressure - Narrow Range	2	1	1, 2	60
7. Drywell Air Temperature	2	1	1, 2	60
8. Drywell Oxygen Concentration - Analyzer and Monitor	2	1	1, 2	62
9. Drywell Hydrogen Concentration - Analyzer and Monitor	2	1	1, 2	62
10. Safety & Relief Valve Position Indicators - Acoustic & Temperature	2/valve (1 each)	1/valve	1, 2	63
11. (Source Range) Neutron Monitors	2	2	1,2	60
12. Drywell Radiation Monitors	2	2	1, 2, 3	61
13. Torus Pressure	2[a]	1	1, 2	60

a This function is shared with Drywell Pressure-Wide Range and Drywell Pressure-Narrow Range.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-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 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 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 62-

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1 restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION 63 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status prior to startup from a COLD SHUTDOWN of longer than 72 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	SA	1, 2
2. Reactor Vessel Water Level	M	SA[d]	1, 2
3 Torus Water Level	M	A	1, 2
4. Torus Water Temperature	M	A	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	Q	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Drywell Hydrogen Concentration - Analyzer and Monitor	M	Q	1, 2
10. Safety/Relief Valve Position Indicators - Acoustic & Temperature	M[c]	E	1, 2
11. (Source Range) Neutron Monitors	M	Q[b]	1, 2, 3
12. Drywell Radiation Monitors	M	E[a]	1, 2
13. Torus Pressure	M	Q	1, 2

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) 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.

(b) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(c) CHANNEL CHECK of the Acoustic Monitors shall consist of verifying the instrument threshold levels.

(d) Analog transmitters are calibrated every 18 months. The control room indicator for the analog transmitter is calibrated at the frequency identified the table.

3.2 - LIMITING CONDITIONS FOR OPERATION

G. Source Range Monitoring

At least the following source range monitor (SRM) channels shall be OPERABLE:

a. In OPERATIONAL MODE 2[a], three.

b. In OPERATIONAL MODE 3 and 4, two.

APPLICABILITY:

OPERATIONAL MODE(s) 2[a], 3, and 4.

ACTION:

1. In OPERATIONAL MODE 2[a] with one of the above required source range monitor CHANNEL(s) inoperable, at least 3 source range monitor CHANNEL(s) shall be restored to OPERABLE status within 4 hours or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

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

4.2 - SURVEILLANCE REQUIREMENTS

G. Source Range Monitoring

Each of the required source range monitor CHANNEL(s) shall be demonstrated OPERABLE by:

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

2. Performance of a CHANNEL CHECK at least once per:

   a. 12 hours in OPERATIONAL MODE 2[a], and

   b. 24 hours in OPERATIONAL MODE(s) 3 or 4.

3. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 7 days prior to startup, and

   b. At least once per 31 days[b].

4. Performance of a CHANNEL CALIBRATION[c] at least once per 18 months[b].

---

a With IRM's on range 2 or below.

b The provisions of Specification 4.0.D are not applicable for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1, provided the surveillance is performed within 12 hours after such entry.

c Neutron detectors may be excluded from the CHANNEL CALIBRATION.

3.2 - LIMITING CONDITIONS FOR OPERATION

H. Explosive Gas Monitoring

The explosive gas monitoring instrumentation CHANNEL(s) shown in Table 3.2.H-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.H are not exceeded.

APPLICABILITY:

During offgas holdup system operation.

ACTION:

1. With an explosive gas monitoring instrumentation CHANNEL alarm/trip setpoint less conservative than required by the above specification, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.H-1.
2. With less than the minimum number of explosive gas monitoring instrumentation CHANNEL(s) OPERABLE, take the ACTION shown in Table 3.2.H-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B to explain why this inoperability was not corrected in a timely manner.
3. The provisions of Specification 3.0.C are not applicable.

4.2 - SURVEILLANCE REQUIREMENTS

H. Explosive Gas Monitoring

Each explosive gas monitoring 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 4.2.H-1.

TABLE 3.2.H-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION

Functional Unit	Minimum CHANNEL(s)	ACTION
MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM		
1. Hydrogen Monitor	1	70

ACTION

ACTION 70 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) OPERABLE requirement, operation of the main condenser offgas treatment system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours. If the recombiner(s) temperature remains constant and THERMAL POWER has not changed, the grab sample collection frequency may be changed to 8 hours.

TABLE 4.2.H-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
Hydrogen Monitor	D	M	Q

3.2 - LIMITING CONDITIONS FOR OPERATION

I. Suppression Chamber and Drywell Spray Actuation

The suppression chamber and drywell spray actuation instrumentation CHANNEL(s) shown in Table 3.2.I-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

With a suppression chamber and drywell spray actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.I-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.I-1.

4.2 - SURVEILLANCE REQUIREMENTS

I. Suppression Chamber and Drywell Spray Actuation

1. Each suppression chamber and drywell spray 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 4.2.I-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM[c]	ACTION
1. Drywell Pressure - High (Permissive)	$0.5 \leq \rho \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level -Low (Permissive)	$\geq$ -48 inches	1	80

ACTION

ACTION 80 -
a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[b] within one hour or declare the suppression chamber and drywell sprays inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the suppression chamber and drywell sprays inoperable.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

c A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains Suppression Chamber and Drywell Spray Actuation capability.

TABLE 4.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Drywell Pressure - High	NA	M	Q
2. Reactor Vessel Water Level -Low	D	M	E[(a)]

---

a Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency indicated in the table.

3.2 - LIMITING CONDITIONS FOR OPERATION

J. Feedwater Pump Trip

The feedwater pump trip instrumentation CHANNEL(s) shown in Table 3.2.J-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.J-1.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

With a feedwater pump trip instrumentation CHANNEL trip setpoint less conservative than value shown in the Trip Setpoint column of Table 3.2.J-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.J-1.

4.2 - SURVEILLANCE REQUIREMENTS

J. Feedwater Pump Trip

1. Each feedwater pump trip 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 4.2.J-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s)[b]	ACTION
Reactor Vessel Water Level -High	≤ 201 inches	2	90

ACTION

ACTION 90 -

a. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum CHANNEL(s) requirement, restore the inoperable CHANNEL to OPERABLE status within 7 days or place the inoperable CHANNEL in the tripped condition within the next 8 hours.

b. With the number of OPERABLE CHANNEL(s) two less than required by the Minimum CHANNEL(s) requirement, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 72 hours or be in at least STARTUP within the next 8 hours.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition.

TABLE 4.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
Reactor Vessel Water Level - High	D	E	E

BASES

3/4.2 INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram (Sections 2.2 and 3/4.1), protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or which terminates operator errors before they result in serious consequences. The objectives of these specifications are to assure the effectiveness of the protective instrumentation when required and to prescribe the trip settings required to assure adequate performance. As indicated, one CHANNEL may be required to be made inoperable for brief intervals to conduct required surveillance. Some of the settings have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations. Surveillance requirements for the instrumentation are selected in order to demonstrate proper function and OPERABILITY. Additional instrumentation for REFUELING operations is identified in Sections 3/4.10.B.

3/4.2.A Isolation Actuation Instrumentation

The isolation actuation instrumentation automatically initiates closure of appropriate isolation valves and/or dampers, which are necessary to prevent or limit the release of fission products from the reactor coolant system, the primary containment and the secondary containment in the event of a loss-of-coolant accident or other reactor coolant pressure boundary (RCPB) leak. The parameters which result in isolation of the secondary containment also actuate the standby gas treatment system. The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of primary and secondary containment and RCPB system isolation. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. Redundant sensor input signals for each parameter are provided for initiation of isolation (one exception is standby liquid control system initiation).

The reactor low level instrumentation is set to trip at greater than or equal to 144 inches above the top of active fuel (which is defined to be 360 inches above vessel zero). Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the loss-of-coolant accident (LOCA) analysis for Dresden Units 2 & 3. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. For this trip setting and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs, even for the maximum break.

BASES

3/4.2.B Emergency Core Cooling System Actuation Instrumentation

The emergency core cooling system (ECCS) instrumentation generates signals to automatically actuate those safety systems which provide adequate core cooling in the event of a design basis transient or accident. The instrumentation which actuates the ECCS is generally arranged in a one-out-of-two taken twice logic circuit. The logic circuit is composed of the four CHANNEL(s) and each CHANNEL contains the logic from the functional unit sensor up to and including all relays which actuate upon a signal from that sensor. For core spray and low pressure coolant injection, the divisionally powered actuation logic is duplicated and the redundant components are powered from the other division's power supply. The single-failure criterion is met through provisions for redundant core cooling functions, e.g., sprays and automatic blowdown and high pressure coolant injection. Although the instruments are listed by system, in some cases the same instrument is used to send the actuation signal to more than one system at the same time.

For effective emergency core cooling during small pipe breaks, the high pressure coolant injection (HPCI) system must function since reactor pressure does not decrease rapidly enough to allow either core spray or the low pressure coolant injection (LPCI) system to operate in time. The automatic pressure relief function is provided as a backup to the HPCI, in the event HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration and also minimizes the risk of inadvertent operation, i.e., only one instrument CHANNEL out-of-service.

3/4.2.C ATWS - RPT Instrumentation

The anticipated transient without scram (ATWS) recirculation pump trip (RPT) provides a means of limiting the consequences of the unlikely occurrence of a failure to scram concurrent with the associated anticipated transient. The response of the plant to this postulated event falls within the bounds of events studied in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO24222, dated December 1979. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases.

3/4.2.D Isolation Condenser Actuation Instrumentation

The isolation condenser system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

BASES

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip of one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s), the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

BASES

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide the operators with the necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas (waste) holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the suppression chamber and drywell spray mode of the low pressure coolant injection/ containment cooling system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

BASES

3/4.2.J Feedwater Trip System Actuation

The feedwater trip system actuation instrumentation is designed to detect a potential failure of the feedwater control system that caused excessive feedwater flow. If undetected, this would lead to reactor vessel water carryover into the main steam lines and to the main turbine. This instrumentation is included in response to Generic Letter 89-19.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.38% $\Delta k/k$ with the highest worth control rod analytically determined, or
2. 0.28% $\Delta k/k$ with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

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

1. By demonstration, prior to or during the first startup after each refueling outage.
2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.
3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall not exceed 1% $\Delta k/k$.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% $\Delta k/k$, 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.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall be verified to be less than or equal to 1% $\Delta k/k$:

1. During the first startup following CORE ALTERATION(s), and
2. At least once per 31 effective full power days.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:
   a. Within one hour:
      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.
      2) Disarm the associated directional control valves[a] either:
         a) Electrically, or
         b) Hydraulically by closing the drive water and exhaust water isolation valves.
   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, 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. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.
2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

---

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

3.3 - LIMITING CONDITIONS FOR OPERATION | 4.3 - SURVEILLANCE REQUIREMENTS

c. Comply with Surveillance Requirement 4.3.A.2 within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

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

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

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

   b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

      1) Electrically, or

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

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

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

3.3 - LIMITING CONDITIONS FOR OPERATION

c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either:

1) Electrically, or

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

3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

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

4.3 - SURVEILLANCE REQUIREMENTS

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

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] 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
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

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

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

APPLICABILITY:

OPERATIONAL MODE(s) 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.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

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

1. Declare the control rods exceeding the above 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. When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

All control rod scram accumulators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2:
   a. With one control rod scram accumulator inoperable, within 8 hours:
      1) Restore the inoperable accumulator to OPERABLE status, or
      2) Declare the control rod associated with the inoperable accumulator inoperable.
   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.
   c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is $\geq$800 psig unless the control rod is fully inserted and disarmed, or scrammed.

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) 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. With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

2) Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:

a) Electrically, or

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

d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.

2. In OPERATIONAL MODE 5[a]:

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

4.3 - SURVEILLANCE REQUIREMENTS

---

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

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

3.3 - LIMITING CONDITIONS FOR OPERATION

1) Electrically, or

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

b. 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.

4.3 - SURVEILLANCE REQUIREMENTS

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:
   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:
      1) Observing any indicated response of the nuclear instrumentation, and
      2) Demonstrating that the control rod will not go to the overtravel position.
   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:
      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by verifying that the control rod drive does not go to the overtravel position:

1. Deleted.
2. Anytime the control rod is withdrawn to the "Full out" position, and
3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

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

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

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

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

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

   b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

      1) Electrically, or

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

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

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

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:
    a. Determine the position of the control rod by an alternate method, or
    b. Move the control rod to a position with an OPERABLE position indicator, or
    c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:
        1) Electrically, or
        2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

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

1. At least once per 24 hours that the position of each control rod is indicated.
2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.
3. Deleted.

---

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

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

   b. Fully insert the control rod.

4.3 - SURVEILLANCE REQUIREMENTS

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

J. Control Rod Drive Housing Support

The control rod drive housing support shall be in place.

APPLICABILITY:

OPERATIONAL MODE(s) 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 at least COLD SHUTDOWN within the following 24 hours.

4.3 - SURVEILLANCE REQUIREMENTS

J. Control Rod Drive Housing Support

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

3.3 - LIMITING CONDITIONS FOR OPERATION

K. SDV Vent and Drain Valves

All scram discharge volume (SDV) vent and drain valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With[b] one or more SDV vent or drain lines with one valve inoperable, isolate[c] the associated line within 7 days or be in HOT SHUTDOWN within the next 12 hours.
2. With[b] one or more SDV vent or drain lines with both valves inoperable, isolate[c] the associated line within 8 hours or be in HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

K. SDV Vent and Drain Valves

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

1. At least once per 31 days by verifying each valve to be open[a], and
2. At least once per 92 days by cycling each valve through at least one complete cycle of travel.
3. At least once per 18 months, the scram discharge volume vent and drain valves shall be demonstrated to:
   a. Close within 30 seconds after receipt of a signal for control rods to scram, and
   b. Open after the scram signal is reset.

---

b Separate Action statement entry is allowed for each SDV vent and drain line.

c An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

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

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER.

ACTION:

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

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.
2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.
3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 20% of RATED THERMAL POWER:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.

---

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

3.3 - LIMITING CONDITIONS FOR OPERATION

M. Rod Block Monitor (RBM)

Both rod block monitor (RBM) CHANNEL(s) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE 1, when thermal power is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

1. With one RBM CHANNEL inoperable:
   a. Verify that the reactor is not operating in a LIMITING CONTROL ROD PATTERN, and
   b. Restore the inoperable RBM CHANNEL to OPERABLE status within 24 hours.
2. With the provisions of ACTION 1 above not met, place the inoperable rod block monitor CHANNEL in the tripped condition within the next one hour.
3. With both RBM CHANNEL(s) inoperable, place at least one inoperable rod block monitor CHANNEL in the tripped condition within one hour.

4.3 - SURVEILLANCE REQUIREMENTS

M. Rod Block Monitor (RBM)

Each of the required RBM CHANNEL(s) shall be demonstrated OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL MODE(s) specified in Table 4.2.E-1.
2. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating in a LIMITING CONTROL ROD PATTERN, but no more often than daily.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

1. Core flow is within 65% to 100% of rated core flow, and
2. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

1. Prior to entry into EGC operation, and
2. At least once per 12 hours while operating in EGC.

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3/4.3.A 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.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

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 during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% Δk/k or R + 0.28% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% Δk/k (or 0.38% Δk/k) margin beyond this condition. 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 following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

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During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by control

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rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.3.D Control Rod Maximum Scram Insertion Times;

3/4.3.E Control Rod Average Scram Insertion Times; and

3/4.3.F Four Control Rod Group Scram Insertion Times

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve

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solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

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3/4.3.G Control Rod Scram Accumulators

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C 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.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

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. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

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3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 4.6.3.5 of the UFSAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

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. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the original SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared

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to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The NRC approved methodology listed in Specification 6.9.A.6 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

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3/4.2.H Source Range Monitoring Function

The source range monitors (SRM) provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. The SRM system performs no automatic safety system function, i.e., it has no scram function. It does provide the operator with a visual indication of neutron level that is needed for knowledgeable and efficient reactor startup at low neutron levels. The consequences of reactivity accidents are functions of the initial neutron flux. The requirements for minimum counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8}% of RATED THERMAL POWER used in the analyses of transients from cold conditions. Two operable SRM channels would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of three operable SRMs is provided as an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3.4 - LIMITING CONDITIONS FOR OPERATION

A. Standby Liquid Control System (SLCS)

The standby liquid control system (SLCS) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2:

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

   b. With both standby liquid control subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 5[a]:

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

   b. With both standby liquid control subsystems inoperable, fully insert all insertable control rods within 1 hour.

4.4 - SURVEILLANCE REQUIREMENTS

A. Standby Liquid Control System

The standby liquid control system shall be demonstrated OPERABLE:

1. At least once per 24 hours by verifying that:

   a. The temperature of the sodium pentaborate solution is greater than or equal to the limits of Figure 3.4.A-1.

   b. The volume of the sodium pentaborate solution is greater than or equal to the limits shown in Figure 3.4.A-2.

   c. The heat tracing circuit is OPERABLE by determining the temperature of the pump suction piping to be greater than or equal to 83°F.

2. At least once per 31 days by:

   a. Verifying the continuity of the explosive charge.

   b. Determining[b] by chemical analysis that the available concentration of boron in solution is 14% by weight to 16.5% by weight.

   c. 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.

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a With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b This surveillance shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits specified by Figure 3.4.A-1.

3.4 - LIMITING CONDITIONS FOR OPERATION

4.4 - SURVEILLANCE REQUIREMENTS

3. When tested pursuant to Specification 4.0.E, by demonstrating that the minimum flow requirement of 40 gpm per pump at a pressure of greater than or equal to 1275 psig is met.

4. At least once per 18 months by:

   a. Initiating one of the standby liquid control subsystems, 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.

   b. Demonstrating that the pump relief valve setpoint is between 1455 and 1545 psig and verifying that the relief valve does not actuate during recirculation to the test tank at normal system pressures.

   c. Demonstrating that the pump suction line from the storage tank is not plugged by manually initiating the system, except the explosive valves, and pumping solution in the recirculation path.

FIGURE 3.4.A-1

SODIUM PENTABORATE SOLUTION TEMPERATURE REQUIREMENTS

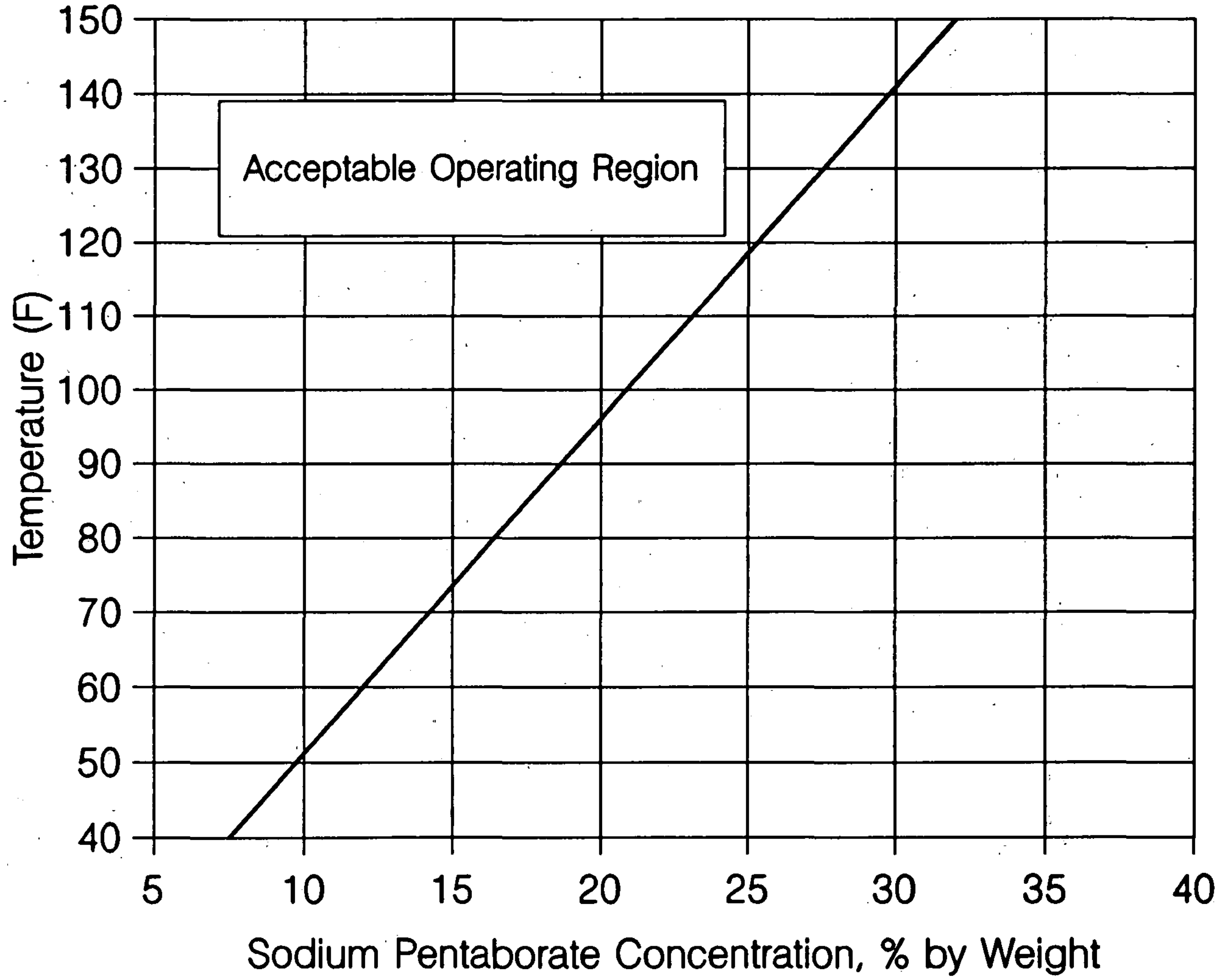
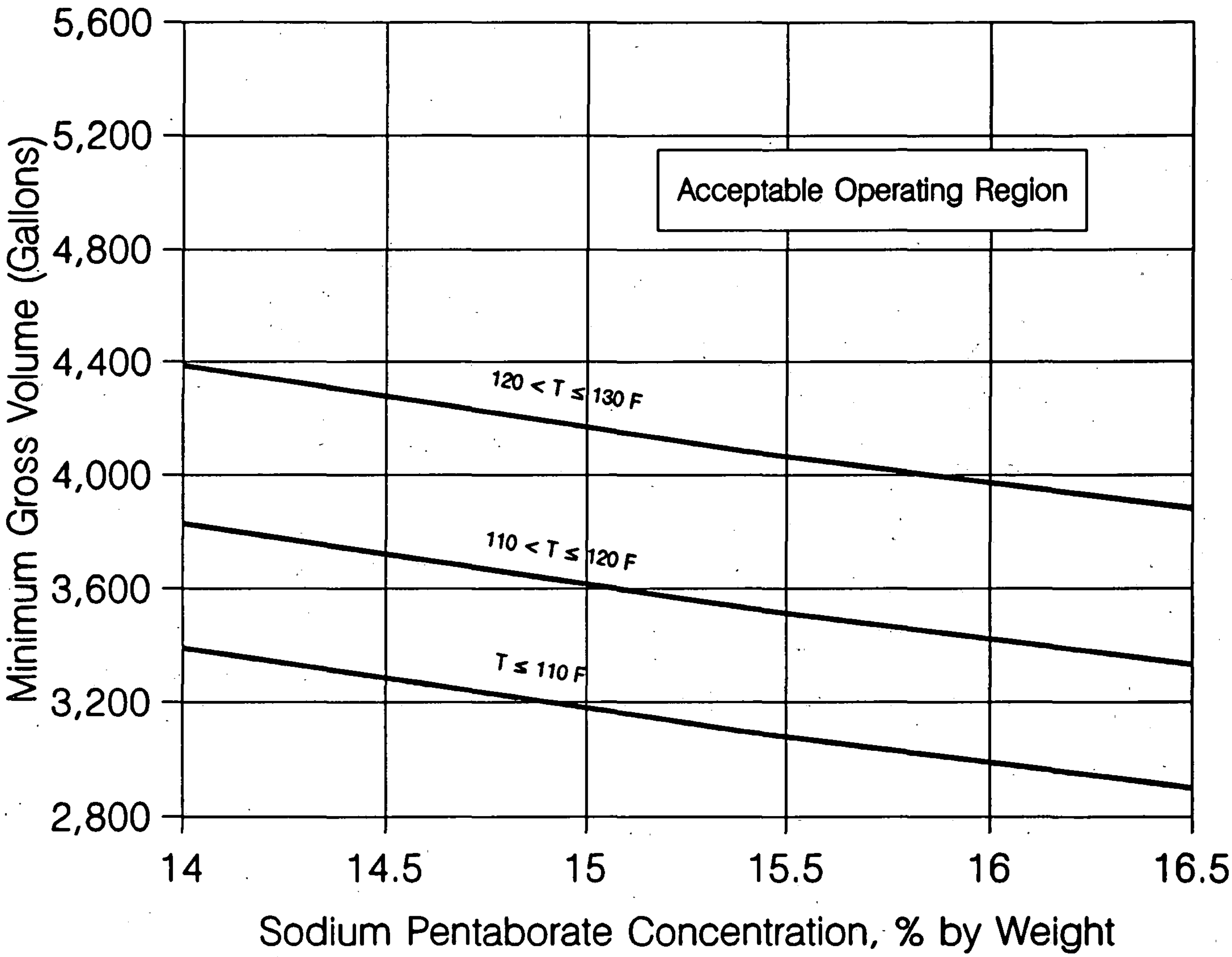


FIGURE 3.4.A-2

SODIUM PENTABORATE SOLUTION VOLUME REQUIREMENTS



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3/4.4.A STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system consists of an unpressurized tank for low temperature sodium pentaborate solution storage, a pair of full capacity positive displacement pumps, two explosive actuated shear plug valves, the poison sparger ring, and the necessary piping, valves and instrumentation. An OPERABLE standby liquid control system provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. OPERABILITY of the system is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the reactor pressure vessel, including the pumps and valves. Two subsystems are required to be OPERABLE; each contains a pump, an explosive valve, and the associated piping, valves, and necessary instruments and controls to ensure an OPERABLE flow path. Inoperability of a nonredundant component, such as the tank, affects both subsystems.

The standby liquid control system provides the capability for bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, it is designed to inject a quantity of boron which produces a concentration of no less than 600 ppm of boron in the reactor core in less than 100 minutes. This boron concentration is required to bring the reactor from full power to 3% $\Delta k/k$ or a more subcritical condition, considering the hot to cold reactivity swing and xenon poisoning. An additional margin of 25% boron is provide to compensate for possible losses and imperfect mixing of the chemical solution in the reactor water. This results in an average concentration of 750 ppm of boron in the reactor core assuming no losses. A net quantity of 3035 gallons of solution at less than or equal to 110°F and having a 14 weight percent sodium pentaborate ($NA_2B_{10}O_{16} \cdot 10H_2O$) concentration is required to meet this shutdown requirement. An additional volume of solution is contained below the pump suction and is not available for injection. Other equivalent combinations of increased concentration and reduced volume are also acceptable provided they have considered required temperatures and net positive suction head.

The specified pumping rate of 40 gpm will meet the above design objective. This insertion rate of boron solution will override the rate of reactivity insertion due to cooldown of the reactor following the xenon peak. Two-pump operation will enable faster reactor shutdown for anticipated transients without scram (ATWS) events. The required minimum flow combined with the solution concentration requirements are sufficient to comply with the requirements of 10 CFR 50.62.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the subsystems inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. The standby liquid control system is operated by a five-position control switch which allows single pump operation for surveillance testing. This testing demonstrates the capability of firing the explosive trigger assemblies, and injects clean demineralized water from the test tank to the reactor vessel to demonstrate the injection line isn't plugged. Locally controlled testing circulates sodium pentaborate from the storage tank, through one suction line, through a pump, and back into the storage tank. This is done separately for each system to demonstrate that both suction lines are not plugged. The only practical time to test the standby liquid control system is during a refueling outage. Components of the system are checked periodically and make a more frequent functional test of the entire system unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the charges are satisfactory. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The two pump operation positions will be used for the injection of the sodium pentaborate into the vessel during an ATWS event. By using the two pump operation position, the standby liquid control system will comply with the requirements of 10 CFR 50.62. Comparison of single-pump test pressures with previous results and correlation of these data with initial two-pump tests are used to verify the capability of the piping.

A 10°F margin is included in Figure 3.4.A-1 above saturation temperature to guard against boron precipitation. Temperature and liquid level alarms for the system are annunciated in the control room. Once the solution has been made up, boron concentration will not vary significantly unless more boron or more water is added. Tank level indication and alarms are provided to indicate whether the solution volume has changed, which might indicate a possible solution concentration change.

Figure 3.4.A-2 provides additional requirements for minimum solution volume, based on existing solution temperature and concentration, to ensure adequate net positive suction head exists for two pump operation. It is permissible to interpolate between the temperature curves.

3.5 - LIMITING CONDITIONS FOR OPERATION

A. Emergency Core Cooling System - Operating

The emergency core cooling systems (ECCS) shall be OPERABLE with:

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

   a. One OPERABLE CS pump, and

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

2. The low pressure coolant injection (LPCI) subsystem comprised of:

   a. Four OPERABLE LPCI pumps, and

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

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

   a. One OPERABLE HPCI pump, and

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

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

4.5 - SURVEILLANCE REQUIREMENTS

A. Emergency Core Cooling System - Operating

The ECCS shall be demonstrated OPERABLE by:

1. At least once per 31 days:

   a. For the CS system, the LPCI subsystem and the HPCI system:

      1) Verifying 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[a] position.

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

2. Verifying that, when tested pursuant to Specification 4.0.E:

   a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of $\geq$90 psig.

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a 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.

3.5 - LIMITING CONDITIONS FOR OPERATION

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:
    a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
    b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. For the LPCI subsystem:
    a. With one LPCI pump inoperable[d], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. Three LPCI pumps together develop a flow of at least 14,500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:
    a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.
    b. For the HPCI system, verifying that:
        1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 350 psig[c].

---

b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A., Action 2.c.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

b. With the LPCI subsystem otherwise inoperable[d], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Isolation Condenser (IC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 150 psig within the following 24 hours.

4. For the ADS:

a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and three LPCI pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE

4.5 - SURVEILLANCE REQUIREMENTS

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the CS and LPCI system discharge line "keep filled" alarm instrumentation.

d. Deleted.

4. At least once per 18 months for the ADS:

a. Verify the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

b. Manually opening each ADS valve when the reactor steam dome pressure is ≥ 150 psig[c] and observing that either:

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

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

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d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A., Action 2.c.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

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 within the following 24 hours.

b. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

5. With an ECCS discharge line "keep filled" pressure alarm instrumentation CHANNEL inoperable, perform Surveillance Requirement 4.5.A.1.a.1) for CS and LPCI at least once per 24 hours.

6. Deleted.

7. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.B 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.

4.5 - SURVEILLANCE REQUIREMENTS

3.5 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling System - Shutdown

At least two of the following four subsystems/loops shall be OPERABLE:

1. One or both core spray (CS) subsystems with:

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

      1) From the suppression chamber, or

      2) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 140,000 available gallons of water.

2. One or both low pressure coolant injection (LPCI) subsystem loops with a subsystem loop comprised of:

   a. At least one OPERABLE LPCI pump, and

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

      1) From the suppression chamber, or

4.5 - SURVEILLANCE REQUIREMENTS

B. Emergency Core Cooling System - Shutdown

The required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.A, except:

1. The LPCI subsystems cross-tie valves may be closed.

2. Each LPCI pump develops the required flow when tested pursuant to Specification 4.0.E.

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

2) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 140,000 available gallons of water.

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5[a].

ACTION:

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

2. With both of the above required subsystems/loops inoperable, suspend CORE ALTERATION(s) and all operations with a potential for draining the reactor vessel. Restore at least one subsystem/loop to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

a The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and water level is maintained within the limits of Specification 3.10.G and 3.10.H.

3.5 - LIMITING CONDITIONS FOR OPERATION

C. Suppression Chamber

The suppression chamber shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2, and 3 with a contained water volume equivalent to a water level of ≥14' 6.5" above the bottom of the suppression chamber.

2. In OPERATIONAL MODE(s) 4 and 5[a] with a contained volume equivalent to a water level of ≥8' above the bottom of the suppression chamber, except that the suppression chamber level may be less than the limit provided that:

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

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

    c. The condensate storage tank contains ≥140,000 available gallons of water, and

    d. The ECCS systems are OPERABLE per Specification 3.5.B.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4 and 5[a].

4.5 - SURVEILLANCE REQUIREMENTS

C. Suppression Chamber

The suppression chamber shall be determined OPERABLE by verifying:

1. For OPERATIONAL MODE(s) 1, 2 and 3, at least once per 24 hours, the water level to be ≥14' 6.5".

2. For OPERATIONAL MODE(s) 4 or 5[a], at least once per 12 hours:

    a. The water level to be ≥8', or

    b. Verify the alternate conditions of Specification 3.5.C.2, or the conditions of footnote (a), to be satisfied.

---

a The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specification 3.10.G and 3.10.H.

3.5 - LIMITING CONDITIONS FOR OPERATION

ACTION:

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

2. In OPERATIONAL MODE(s) 4 or 5[a] with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATION(s) 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.

4.5 - SURVEILLANCE REQUIREMENTS

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a The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specification 3.10.G and 3.10.H.

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Isolation Condenser

The isolation condenser (IC) system shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure > 150 psig.

ACTION:

With the IC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the IC 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 within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser

The IC system shall be demonstrated OPERABLE:

1. At least once per 24 hours by verifying the shell side water volume and the shell side water temperature to be within limits.
2. 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.
3. At least once per 18 months by verifying the IC system actuates on an actual or simulated automatic initiation signal.
4. At least once per 5 years by verifying the system heat removal capability.

BASES

3/4.5.A ECCS - Operating

3/4.5.B ECCS - Shutdown

The Core Spray (CS) system, together with the Low Pressure Coolant Injection (LPCI) subsystem, is provided to assure that the core is adequately cooled following a loss-of-coolant accident (LOCA) 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 Automatic Depressurization System (ADS).

The CS system is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the CS 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 system functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The Low Pressure Coolant Injection (LPCI) subsystem is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The LPCI subsystem with a minimum of three pumps will 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 subsystem 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 system functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The High Pressure Coolant Injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shutdown while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which CS operation or LPCI subsystem operation maintains adequate core cooling.

The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 5000 gpm at steam supply pressures between 1150 and 150 psig. Suction piping for the system is provided from the condensate storage tank and the suppression pool. Pump suction for HPCI is normally aligned to the condensate storage tank source to minimize injection of suppression pool water into the reactor vessel. However, if the condensate storage tank water supply is low, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI system.

BASES

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified Automatic Depressurization System and both the CS system and LPCI subsystem. In addition, the Isolation Condenser (IC) system, a system for which no credit is taken in the safety analysis, will automatically initiate on a sustained reactor high pressure condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the IC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete system functional test requires a reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant, the Automatic Depressurization System (ADS) automatically causes all OPERABLE main steamline relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls the five main steamline relief valves although safety analyses support a minimum of 4 OPERABLE valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability. A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the ADS discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured steam flow or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test once only. The pressure specified for this test is that pressure recommended by the valve manufacturer. Reactor startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Thus, a footnote is included in this SR to indicate that 4.0.D does not apply.

To preserve single failure criteria, a minimum of two independent OPERABLE low-pressure ECCS subsystems/loops are required in OPERATIONAL MODE(s) 4 and 5 to ensure adequate vessel inventory makeup in the event of an inadvertent vessel draindown. Only a single LPCI pump is required per loop because of the large injection capacity. All of the ECCS may be inoperable provided the reactor head is removed, the reactor cavity is flooded, the spent fuel gates are removed, and the water level is maintained within the limits required by the Refueling Operations specifications.

BASES

3/4.5.C Suppression Chamber

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI and CS systems and the LPCI subsystem in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL MODE(s) 1, 2 or 3 is also required by Specification 3.7.K.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and concurrently provide assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL MODE(s) 4 or 5.

In OPERATIONAL MODE(s) 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 212°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on net positive suction head (NPSH), recirculation volume and vortex prevention plus a safety margin for conservatism. With the suppression chamber water level less than the required limit, all ECCS subsystems are inoperable unless they are aligned to an OPERABLE condensate storage tank. When the suppression chamber level is less than 8 feet, the CS system or the LPCI subsystem is considered OPERABLE only if it can take suction from the condensate storage tank, and the condensate storage tank water level is sufficient to provide the required NPSH for the CS or LPCI pumps. Therefore, a verification that either the suppression chamber water level is greater than or equal to 8 feet or that CS or LPCI is aligned to take suction from the condensate storage tank and the condensate storage tank contains greater than or equal to 140,000 gallons of water, ensures CS or LPCI can supply at least 50,000 gallons of make-up water to the reactor pressure vessel. The CS suction is uncovered at the 90,000 gallon level.

3/4.5.D Isolation Condenser

The isolation condenser is provided for core decay heat removal following reactor isolation from the main condenser and reactor scram. The isolation condenser has a heat removal capacity (252.5 x 10^6 BTU/hour) sufficient to handle the decay heat production at 300 seconds following a scram. Following a reactor scram and an isolation from the main condenser, water will be lost from the reactor vessel through the relief valves during the first 300 seconds. This represents a minor loss relative to the vessel inventory.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1070 psig sustained for 17 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered OPERABLE, the shell side of the isolation condenser must contain at least 20,000 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the

BASES

condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are OPERABLE from on-site power. The preferred source of make-up water for the Isolation Condenser is the clean demineralized water system. The fire protection system is also available as make-up water.

3.6 - LIMITING CONDITIONS FOR OPERATION

A. Recirculation Loops

Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With only one reactor coolant system recirculation loop in operation, within 24 hours either, restore both loops to operation or:

   a. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.B, and

   b. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Operating Limit by 0.01 per Specification 3.11.C, and

   c. Reduce the Average Power Range Monitor (APRM) Flow Biased Neutron Flux Scram and Rod Block and Rod Block Monitor Trip Setpoints to those applicable to single recirculation loop operation per Specifications 2.2.A and 3.2.E.

   d. Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) to single loop operation limits as specified in the CORE OPERATING LIMITS REPORT (COLR).

4.6 - SURVEILLANCE REQUIREMENTS

A. Recirculation Loops

Each pump motor generator (MG) set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with the overspeed setpoints specified in the CORE OPERATING LIMITS REPORT at least once per 18 months.

3.6 - LIMITING CONDITIONS FOR OPERATION

e. Electrically prohibit the idle recirculation pump from starting[a].

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

2. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least STARTUP within 8 hours and in HOT SHUTDOWN within the next 6 hours.

4.6 - SURVEILLANCE REQUIREMENTS

a Except to permit testing in preparation for returning the pump to service.

3.6 - LIMITING CONDITIONS FOR OPERATION

B. Jet Pumps

All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least 19 jet pumps.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one or more jet pumps inoperable for other than inoperable flow indication, be in at least HOT SHUTDOWN within 12 hours.
2. With flow indication inoperable for two or more jet pumps, flow indication shall be restored such that at least 19 jet pumps have OPERABLE flow indication within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows:

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verifying that no two of the following conditions occur when both recirculation pumps are operating in accordance with Specification 3.6.C:
   a. The indicated recirculation pump flow differs by > 10% from the established speed-flow characteristics.
   b. The indicated total core flow differs by > 10% from the established total core flow value derived from established core plate ΔP/core flow relationships.
   c. The indicated flow of any individual jet pump differs from the established patterns by > 10%.
   d. The provisions of Specification 4.0.D are not applicable provided that the surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

2. During single recirculation loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by verifying that no two of the following conditions occur:

   a. The indicated recirculation pump flow in the operating loop differs by >10% from the established single recirculation speed-flow characteristics.

   b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.

   c. The indicated flow of any individual jet pump differs from established single recirculation loop patterns by >10%.

   d. The provisions of Specification 4.0.D are not applicable provided that the surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

3.6 - LIMITING CONDITIONS FOR OPERATION

C. Recirculation Pumps

Recirculation pump speed shall be maintained within:

1. 10% of each other with THERMAL POWER ≥80% of RATED THERMAL POWER.

2. 15% of each other with THERMAL POWER <80% of RATED THERMAL POWER.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2 during two recirculation loop operation.

ACTION:

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

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

2. Trip one of the recirculation pumps and take the ACTION required by Specification 3.6.A.1.

4.6 - SURVEILLANCE REQUIREMENTS

C. Recirculation Pumps

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

3.6 - LIMITING CONDITIONS FOR OPERATION

D. Idle Recirculation Loop Startup

An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel and the bottom head coolant temperature is within limits[a], and:

1. 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 within limits, or

2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is within limits.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any recirculation loop, restore the parameter(s) to within limits within 30 minutes, and determine if the reactor coolant system is acceptable for continued operation within 72 hours.

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

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup

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

a Below 25 psig reactor pressure, this temperature differential is not applicable.

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

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

2. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. Deleted.

2. At least once per 18 months, 1/2 of the safety valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations. At least once per 40 months, the safety valves shall be rotated such that all 9 safety valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations.

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

b Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function
Setpoint (psig)

Open
≤ 1112 psig
≤ 1112 psig
≤ 1135 psig
≤ 1135 psig
≤ 1135 psig[a]

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more relief valves open, provided that suppression pool average water temperature is <110°F, take action to close the open relief valve(s); if suppression pool average water temperature is ≥110°F place the reactor mode switch in the Shutdown position.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL FUNCTIONAL TEST of the relief valve function at least once per 18 months, and a
   b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST of the entire system at least once per 18 months.
2. Deleted.

a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

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

3. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

G. Leakage Detection Systems

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

1. The primary containment atmosphere particulate radioactivity sampling system, and
2. The drywell floor drain sump system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the primary containment atmosphere particulate radioactivity sampling system inoperable, restore the inoperable leak detection radioactivity sampling system to OPERABLE status within 24 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With the drywell floor drain sump system inoperable, restore the drywell floor drain sump system to OPERABLE status within 24 hours; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

G. Leakage Detection Systems

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

1. Performing the leakage determinations of Specification 4.6.H.
2. Performing a CHANNEL CALIBRATION of the drywell floor drain sump pump discharge flow integrator at least once per 18 months.

3.6 - LIMITING CONDITIONS FOR OPERATION

H. Operational Leakage

Reactor coolant system leakage shall be limited to:

1. No PRESSURE BOUNDARY LEAKAGE.
2. ≤25 gpm total leakage averaged over any 24 hour surveillance period.
3. ≤5 gpm UNIDENTIFIED LEAKAGE.
4. ≤2 gpm increase in UNIDENTIFIED LEAKAGE within any period of 24 hours or less (Applicable in OPERATIONAL MODE 1 only).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
2. With the reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) greater than the above limit(s), 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.

4.6 - SURVEILLANCE REQUIREMENTS

H. Operational Leakage

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

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a], and
2. Determining the primary containment sump flow rate at least once per 8 hours, not to exceed 12 hours.

a Not a means of quantifying leakage.

3.6 - LIMITING CONDITIONS FOR OPERATION

3. With an increase in reactor coolant system UNIDENTIFIED LEAKAGE of >2 gpm within any period of 24 hours or less in OPERATIONAL MODE 1:

   a. Identify the source of leakage as not IGSCC susceptible material within 4 hours, or

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

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

I. Chemistry

The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.6.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

1. In OPERATIONAL MODE 1:

   a. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1;

      1) For ≤72 hours during one continuous time interval, and

      2) For ≤336 hours per year for conductivity and chloride concentration, and

      3) With the conductivity ≤10 μmho/cm at 25°C and with the chloride concentration ≤0.5 ppm,

      the condition does not need to be reported to the Commission.

   b. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1;

      1) For >72 hours during one continuous time interval, or

4.6 - SURVEILLANCE REQUIREMENTS

I. Chemistry

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

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

2. Analyzing a sample of the reactor coolant for:

   a. Chlorides at least once per:

      1) 72 hours, and

      2) 8 hours whenever conductivity is greater than the limit in Table 3.6.I-1.

   b. Conductivity at least once per 72 hours.

   c. pH at least once per 8 hours whenever conductivity is greater than the limit in Table 3.6.I-1.

3. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, obtaining an in-line conductivity measurement at least once per 4 hours.

---

a The provisions of Specification 3.0.D are not applicable during unit shutdown when entering OPERATIONAL MODE(s) 2 and 3 from OPERATIONAL MODE 1.

3.6 - LIMITING CONDITIONS FOR OPERATION

2) For >336 hours per year for conductivity and chloride concentration,

Be in at least STARTUP within the next 8 hours.

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

2. In OPERATIONAL MODE(s) 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1 for >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.

4.6 - SURVEILLANCE REQUIREMENTS

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

   a. 7 days, and

   b. 24 hours whenever conductivity is greater than the limit in Table 3.6.I-1.

TABLE 3.6.I-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL MODE(s)	Chlorides	Conductivity (μmhos/cm @25°C)	pH
1	≤0.2 ppm	≤1.0	5.6≤ pH ≤8.6
2 and 3	≤0.1 ppm	≤2.0	5.6≤ pH ≤8.6

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3, with any main steam line not isolated.

ACTION:

1. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours and restore DOSE EQUIVALENT I-131 to within limits within 48[a] hours

2. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131, for greater than 48 hours or with specific activity of the reactor coolant >4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours, and isolate all main steam lines within 12 hours, or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

In OPERATIONAL MODE 1 the specific activity of the reactor coolant shall be verfied to be ≤ 0.2 μCi/gram DOSE EQUIVALENT I-131 once per 7 days.

a The provisions of Specification 3.0.D are not applicable.

3.6 - LIMITING CONDITION FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENT

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3.6 - LIMITING CONDITION FOR OPERATION 4.6 - SURVEILLANCE REQUIREMENT

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3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.6.K-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

1. A maximum reactor coolant heatup of 100°F in any one hour period,
2. A maximum reactor coolant cooldown of 100°F in any one hour period,
3. A maximum reactor coolant temperature change of $\leq$20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
4. The reactor vessel flange and head flange temperature $\geq$100°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

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 required heatup and cooldown limits and to the right of the limit lines of Figure 3.6.K-1 curves A, or B, as applicable, at least once per 30 minutes.
2. The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.6.K-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.
3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.
4. The reactor vessel flange and head flange temperature shall be verified to be $\geq$100°F:
   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:
      1) $\leq$130°F, at least once per 12 hours.
      2) $\leq$110°F, at least once per 30 minutes.
   b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

3.6 - LIMITING CONDITIONS FOR OPERATION

ACTION:

With any of the above limits exceeded,

1. Restore the temperature and/or pressure to within the limits within 30 minutes, and

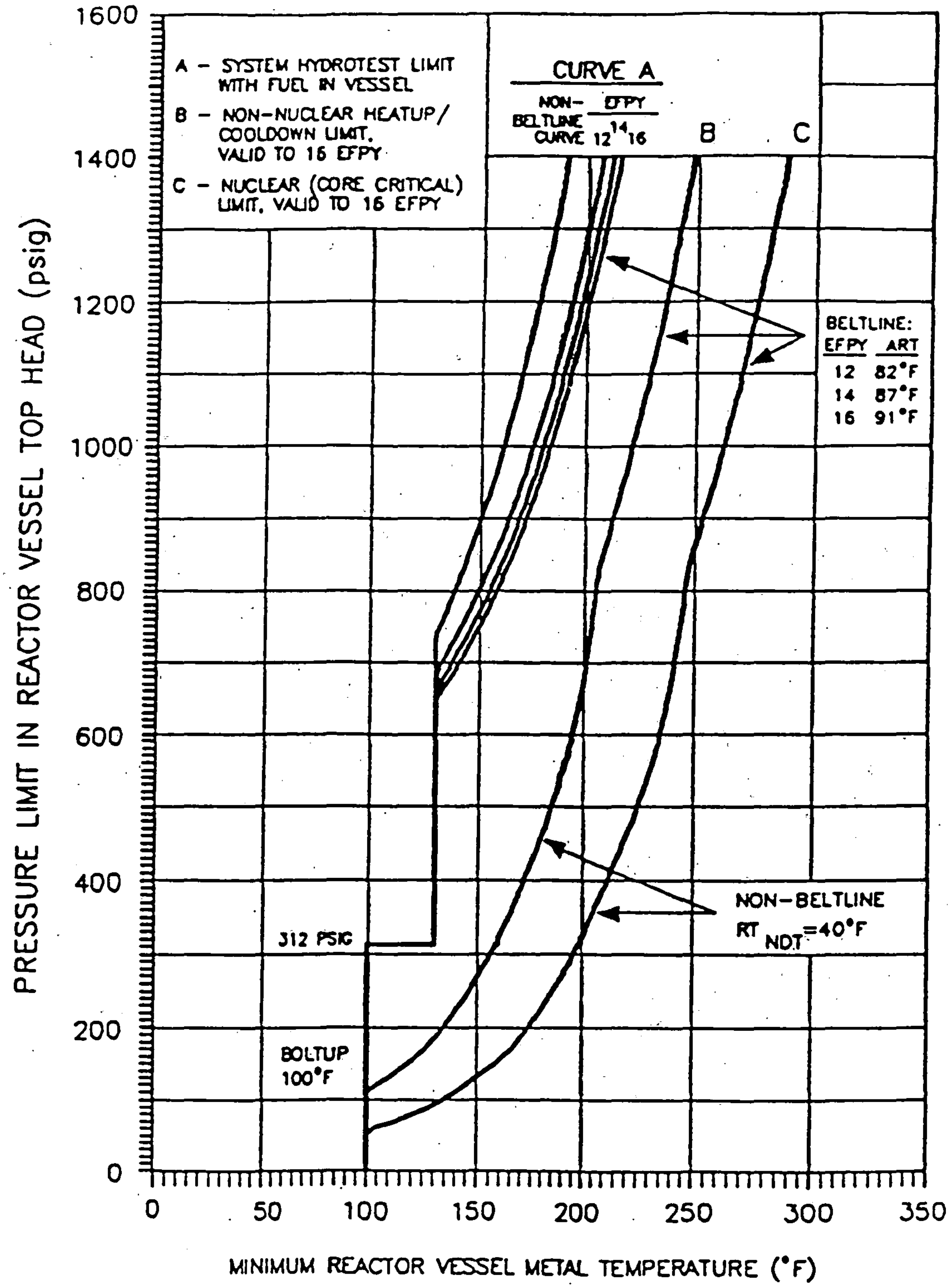
2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations, or

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

FIGURE 3.6.K-1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE



3.6 - LIMITING CONDITIONS FOR OPERATION

L. Reactor Steam Dome Pressure

The pressure in the reactor steam dome shall be ≤1005 psig.

APPLICABILITY:

OPERATIONAL MODE(s) 1[a] and 2[a]

ACTION:

With the reactor steam dome pressure >1005 psig, reduce the pressure to ≤1005 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

4.6 - SURVEILLANCE REQUIREMENTS

L. Reactor Steam Dome Pressure

The reactor steam dome pressure shall be verified to be ≤1005 psig at least once per 12 hours.

a Not applicable during anticipated transients.

3.6 - LIMITING CONDITIONS FOR OPERATION

M. Main Steam Line Isolation Valves

Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times ≥3 seconds and ≤5 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

With one or more MSIVs inoperable, maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours either:

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

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

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

4.6 - SURVEILLANCE REQUIREMENTS

M. Main Steam Line Isolation Valves

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.E.

3.6 - LIMITING CONDITIONS FOR OPERATION

N. Structural Integrity

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.6.N.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4 and 5.

ACTION:

1. 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 limits or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.
2. 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).
3. 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.

4.6 - SURVEILLANCE REQUIREMENTS

N. Structural Integrity

No additional Surveillance Requirements other than those required by Specification 4.0.E.

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Shutdown Cooling - HOT SHUTDOWN

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b,c], with each loop consisting of at least:

1. One OPERABLE SDC pump, and
2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive setpoint.

ACTION:

1. With less than the above required SDC 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 SDC loop. Be in at least COLD SHUTDOWN within 24 hours[d].

4.6 - SURVEILLANCE REQUIREMENTS

O. Shutdown Cooling - HOT SHUTDOWN

At least one SDC loop, one recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

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

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

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

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

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With no SDC loop or recirculation pump in operation, immediately initiate corrective action to return at least one shutdown cooling loop or recirculation pump 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.

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Shutdown Cooling - COLD SHUTDOWN

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b][c] with each loop consisting of at least:

1. One OPERABLE SDC pump, and
2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION:

1. With less than the above required SDC 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 SDC loop.
2. With no SDC loop or recirculation pump 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.

4.6 - SURVEILLANCE REQUIREMENTS

P. Shutdown Cooling - COLD SHUTDOWN

At least one SDC loop, recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

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

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

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

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3/4.6.A Recirculation Loops

3/4.6.B Jet Pumps

3/4.6.C Recirculation Pumps

3/4.6.D Idle Recirculation Loop Startup

The reactor coolant recirculation system is designed to provide a forced coolant flow through the core to remove heat from the fuel. The reactor coolant recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. The operation of the reactor coolant recirculation system is an initial condition assumed in the design basis loss-of-coolant accident (LOCA). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The analyses assumes both loops are operating at the same flow prior to the accident. If a LOCA occurs with a flow mismatch between the two loops, the analysis conservatively assumes the pipe break is in the loop with the higher flow.

A plant specific analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that in the event of a LOCA caused by a pipe break in the operating recirculation loop, the ECCS response will provide adequate core cooling. The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR fuel cladding integrity Safety Limit is increased as noted by Specification 2.1.B. The Reactor Protection System APRM scram and control rod block setpoints are also required to be adjusted to account for the different response of the reactor and different relationships between recirculation drive flow and reactor core flow. During single loop operation for greater than 24 hours, the idle recirculation pump is electrically prohibited from starting until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Jet pump OPERABILITY is an explicit assumption in the design basis LOCA analysis. The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If a beam holding a jet pump in place fails, the jet pump suction and mixer sections could become displaced, resulting in a larger flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

The surveillance requirements for jet pumps are designed to detect a significant degradation in jet pump performance that precedes a jet pump failure. Significant degradation is indicated if more than one of the three specified criteria confirms unacceptable deviations from established patterns or relationships. A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation. The agreement of indicated core plate dp and core flow relationships provides

BASES

assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal pattern provides the indication necessary to detect a failed jet pump. Allowable deviations from the established patterns have been developed based on operation. Since refueling activities (fuel assembly replacement or shuffle, as well as any modification to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be reestablished each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate jet pump failure.

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within the limit specified in the Dresden Administrative Technical Requirements prior to startup of an idle loop. The loop temperature must also be within the limit specified in the Dresden Administrative Technical Requirements to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than the limit specified in the Dresden Administrative Technical Requirements. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of the rated pump speed limit specified in the Dresden Administrative Technical Requirements for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

In addition to suspending startup of an idle recirculation loop not meeting the temperature limits, the temperature parameters must be restored within 30 minutes. The 30 minute completion time

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reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if operation can continue. The evaluation must verify the reactor coolant system integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour completion time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

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Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

3/4.6.G 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. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates. Leakage from the reactor coolant pressure boundary inside the drywell is detected by at least one or two independently monitored variables, such as sump level changes and drywell atmosphere radioactivity levels. The means of quantifying leakage in the drywell is the drywell floor drain sump pumps. With the drywell floor drain sump pump system inoperable, no other form of monitoring can provide the equivalent information. However, primary containment atmosphere sampling for radioactivity can provide indication of changes in leakage rates.

3/4.6.H Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

An UNIDENTIFIED LEAKAGE increase of more than 2 gpm within a 24 hour period is an indication of a potential flaw in the reactor coolant pressure boundary and must be quickly evaluated. Although the increase does not necessarily violate the absolute UNIDENTIFIED LEAKAGE limit, IGSCC susceptible components must be determined not to be the source of the leakage within the required completion time.

3/4.6.I 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

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concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during power operation.

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.

Action 1 permits temporary operation with chemistry limits outside of the limits required in OPERATIONAL MODE 1 without requiring Commission notification. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4 of the FSAR. During startup and

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shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The pressure-temperature limit lines shown in Figure 3.6.K-1, for operating conditions; Inservice Hydrostatic Testing (curve A), Non-Nuclear Heatup/Cooldown (curve B), and Core Critical Operation (curve C). The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it is treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

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Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a

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vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.6.K-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M 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.

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3/4.6.N 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.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Shutdown Cooling - HOT SHUTDOWN

3/4.6.P Shutdown Cooling - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

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.

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] 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.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. Perform required visual examinations and leakage rate testing except for primary containment air lock testing in accordance with and at the frequency specified by the Primary Containment Leakage Rate Testing Program.
2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed, except for valves that are open under administrative control as permitted by Specification 3.7.D.
3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.
4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. 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 per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

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4.7 - SURVEILLANCE REQUIREMENTS

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3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

1. With one primary containment air lock door inoperable:

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

   b. 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[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By performing required primary containment air lock leakage testing in accordance with the Primary Containment Leakage Rate Testing Program[c][d].

2. At least once per 6 months, by verifying that only one door in each air lock can be opened at a time[e].

---

a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

d Results shall be evaluated against acceptance criteria applicable to Specification 4.7.A.1.

e Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.

3.7 - LIMITING CONDITIONS FOR OPERATION

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

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not opened simultaneously.

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, 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.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

D. Primary Containment Isolation Valves

Each primary containment isolation valve and reactor instrumentation excess flow check valve shall be OPERABLE[a].

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more of the primary containment isolation valve(s)[b] inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

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

    b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position[a], or

    c. Isolate each affected penetration by use of at least one closed manual valve or blind flange[a].

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

4.7 - SURVEILLANCE REQUIREMENTS

D. Primary Containment Isolation Valves

1. Each power-operated or automatic primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

2. Each power-operated or automatic primary containment isolation valve required to close on an isolation signal, except traversing in-core probe system explosive isolation valves, shall be demonstrated OPERABLE at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

3. The isolation time of each power-operated or automatic primary containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.E.

4. Each reactor instrumentation line excess flow check valve which fulfills a primary containment isolation function shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

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

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a Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

b Except main steam isolation valves (MSIVs). Required actions for inoperable MSIVs are provided in Specification 3.6.M.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. 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.

4.7 - SURVEILLANCE REQUIREMENTS

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

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is ≤ 11.5 scfh.

3.7 - LIMITING CONDITIONS FOR OPERATION

E. Suppression Chamber - Drywell Vacuum Breakers

Nine suppression chamber - drywell vacuum breakers shall be OPERABLE and twelve suppression chamber - drywell vacuum breakers shall be closed.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more of the required suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore at least nine 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.

2. With one suppression chamber - drywell vacuum breaker open, restore the open vacuum breaker to the closed position within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one position indicator of any OPERABLE suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or visually verify the vacuum breaker to be closed at least once per 24 hours. Otherwise, declare the vacuum breaker inoperable.

4.7 - SURVEILLANCE REQUIREMENTS

E. Suppression Chamber - Drywell Vacuum Breakers

Each suppression chamber - drywell vacuum breaker shall be:

1. Verified closed at least once per 7 days.

2. Demonstrated OPERABLE:

   a. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from one or more main steam relief valve(s), by cycling each vacuum breaker through at least one complete cycle of full travel.

   b. At least once per 31 days by verifying both position indicator(s) OPERABLE by observing expected valve movement during the cycling test.

   c. At least once per 18 months by:

      1) Verifying the force required to open the vacuum breaker, from the closed position, to be ≤0.5 psid, and

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

      3) Verifying that each valve's position indicator is capable of detecting disk displacement of ≥0.0625 inches.

3.7 - LIMITING CONDITIONS FOR OPERATION

F. Reactor Building - Suppression Chamber Vacuum Breakers

All reactor building - suppression chamber vacuum breakers shall be OPERABLE and closed.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one reactor building - suppression chamber vacuum breaker line inoperable for opening with both valves known to be closed, restore the inoperable vacuum breaker line to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one reactor building - suppression chamber vacuum breaker line otherwise inoperable, verify at least one vacuum breaker in the line to be closed within 2 hours and restore the open vacuum breaker to the closed position within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the position indicator of the air operated reactor building - suppression chamber vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker to be closed at least once per 24 hours by an

4.7 - SURVEILLANCE REQUIREMENTS

F. Reactor Building - Suppression Chamber Vacuum Breakers

Each reactor building - suppression chamber vacuum breaker shall be:

1. Verified closed at least once per 7 days.

2. Demonstrated OPERABLE:

   a. At least once per 92 days when tested pursuant to Specification 4.0.E by:

      1) Cycling the vacuum breaker through at least one test cycle.

      2) Verifying the air operated vacuum breaker position indicator OPERABLE by observing expected valve movement during the cycling test.

   b. At least once per 18 months by:

      1) Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of 0.5 psid.

      2) Verifying the air operated vacuum breaker position indicator OPERABLE by performance of a CHANNEL CALIBRATION.

3.7 - LIMITING CONDITIONS FOR OPERATION

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

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

G. Drywell Internal Pressure

The drywell internal pressure shall not exceed +1.5 psig[a].

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the drywell internal pressure <1.0 psig during the applicable time period for OPERATIONAL MODE 1, restore the internal pressure to above the low pressure limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

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

4.7 - SURVEILLANCE REQUIREMENTS

G. Drywell Internal Pressure

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

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a In OPERATIONAL MODE 1, during the time period beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown, the drywell internal pressure shall also be maintained ≥1.0 psig (except for up to 4 hours for required surveillance which reduces the differential pressure.)

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be ≥1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to ≤15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.
2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.
2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL CHECK at least once per 24 hours,
   b. CHANNEL CALIBRATION at least once every 31 days.

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a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With the drywell and/or suppression chamber pressure instrumentation CHANNEL(s) inoperable, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or reduce THERMAL POWER to $<15\%$ RATED THERMAL POWER within the next 8 hours.

4. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable and with insufficient drywell and suppression chamber pressure instrumentation CHANNEL(s) OPERABLE to determine drywell - suppression chamber differential pressure, restore either the drywell - suppression chamber differential pressure instrumentation CHANNEL or sufficient drywell and suppression chamber pressure instrumentation CHANNEL(s) to determine drywell - suppression chamber differential pressure to OPERABLE status within 8 hours or reduce THERMAL POWER to $<15\%$ RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION	4.7 - SURVEILLANCE REQUIREMENTS
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3.7 - LIMITING CONDITIONS FOR OPERATION

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell atmosphere oxygen concentration shall be <4% by volume.

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER and at least once per 7 days thereafter.

3.7 - LIMITING CONDITIONS FOR OPERATION

K. Suppression Chamber

The suppression chamber shall be OPERABLE with:

1. The suppression pool water level between 14' 6.5" and 14' 10.5",

2. A suppression pool maximum average water temperature of ≤95°F during OPERATIONAL MODE(s) 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a. ≤105°F during testing which adds heat to the suppression pool.

   b. ≤110°F with THERMAL POWER ≤1% of RATED THERMAL POWER.

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

3. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1 inch diameter orifice at a differential pressure of 1.0 psid.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the suppression pool water level outside the above limits, restore the water level to within the limits

4.7 - SURVEILLANCE REQUIREMENTS

K. Suppression Chamber

The suppression chamber shall be demonstrated OPERABLE:

1. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

2. At least once per 24 hours by verifying the suppression pool average water temperature to be ≤95°F, except:

   a. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature to be ≤105°F.

   b. At least once per hour when suppression pool average water temperature is ≥ 95°F, by verifying:

      1) Suppression pool average water temperature to be ≤ 110°F, and

      2) THERMAL POWER to be ≤ 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 95°F for more than 24 hours.

   c. At least once per 30 minutes with the main steam isolation valves closed following a scram and suppression pool average water temperature >95°F, by verifying suppression pool average water temperature to be ≤120°F.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. Deleted.

4. Deleted.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. 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.

3.7 - LIMITING CONDITIONS FOR OPERATION

L. Suppression Chamber and Drywell Spray

The suppression chamber and drywell spray functions of the low pressure coolant injection (LPCI)/containment cooling system shall be OPERABLE with two independent loops, each loop consisting of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of recirculating water from the suppression pool through a heat exchanger and the suppression chamber and drywell spray nozzles.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one suppression chamber/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.

2. With both suppression chamber/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 next 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

L. Suppression Chamber and Drywell Spray

The suppression chamber and drywell spray functions of LPCI/containment cooling system shall be demonstrated OPERABLE:

1. 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.

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

3.7 - LIMITING CONDITIONS FOR OPERATION

M. Suppression Pool Cooling

The suppression pool cooling function of the low pressure coolant injection (LPCI)/containment cooling system shall be OPERABLE with two independent loops, each loop consisting of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of recirculating water from the suppression pool through a heat exchanger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one suppression pool cooling 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.

2. 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.

4.7 - SURVEILLANCE REQUIREMENTS

M. Suppression Pool Cooling

The suppression pool cooling function of the LPCI/containment cooling system shall be demonstrated OPERABLE:

1. 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.

2. By verifying that each of the required LPCI pumps develops the required recirculation flow through the heat exchanger and the suppression pool when tested pursuant to Specification 4.0.E.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 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.
2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

1. Verifying at least once per 24 hours that the pressure within the secondary containment is $\geq$0.25 inches of vacuum water gauge.
2. Verifying at least once per 31 days that:
   a. At least one door in each secondary containment air lock is closed.
   b. All secondary containment penetrations[a] not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed.
3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate $\leq$4000 cfm for one hour and maintaining $\geq$0.25 inches of vacuum water gauge in the secondary containment.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally locked or sealed-closed penetrations may be opened intermittently under administrative controls.

3.7 - LIMITING CONDITIONS FOR OPERATION

O. Secondary Containment Automatic Isolation Dampers

Each secondary containment ventilation system automatic isolation damper shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

1. Restore the inoperable damper(s) to OPERABLE status, or
2. Isolate each affected penetration by use of at least one deactivated automatic damper secured in the isolation position, or
3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL MODE(s) 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 MODE *, suspend handling of irradiated fuel in the secondary containment, CORE

4.7 - SURVEILLANCE REQUIREMENTS

O. Secondary Containment Automatic Isolation Dampers

Each secondary containment ventilation system automatic isolation damper shall be demonstrated OPERABLE:

1. Prior to returning the damper to service after maintenance, repair, or replacement work is performed on the valve/damper or its associated actuator, control, or power circuit by performance of a cycling test.
2. At least once per 18 months by verifying that on an isolation test signal each automatic isolation damper actuates to its isolation position.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

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

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

2. With both standby gas treatment subsystems otherwise inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. 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 operating.

2. 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:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ±10%.

   b. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% 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%.

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3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting leakage to less than 1.0 La except prior to the first startup after performing a required leakage test. At this time, Primary Containment Leakage Rate Testing Program leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (La) is 1.6% by weight of the containment atmosphere per day at the design basis LOCA maximum peak containment pressure (Pa) of 48 psig.

Surveillance requirements maintain PRIMARY CONTAINMENT INTEGRITY by requiring compliance with visual examinations and leakage rate test requirements and test frequencies of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing or main steam isolation valve leakage does not necessarily result in a failure of the surveillance requirement. The impact of the failure to meet such surveillance requirements must be evaluated against the Type A, Type B and Type C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be less than 0.60 La for combined Type B and Type C leakage, and 0.75 La for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of less than or equal to 1.0 La. At less than or equal to 1.0 La, the off-site dose consequences are bounded by the assumptions of the safety analysis.

3/4.7.B DELETED

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specification 3.7.A. 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.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. The surveillance requirements reflect the leakage rate testing requirements with respect to air lock leakage (Type B

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leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. The surveillance requirements have been annotated such that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and Type C primary containment leakage.

3/4.7.D 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. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. G. Eisenhut (NRC) to Mr. L. DelGeorge (CECo) dated June 25, 1982.)

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers

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are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

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 is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

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3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I DELETED

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup

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and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed 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.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation 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. 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 or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

<u>3/4.7.L</u> <u>Suppression Chamber and Drywell Spray</u>

Following a Design Basis Accident (DBA), the suppression chamber spray function of the low pressure coolant injection (LPCI)/containment cooling system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the LPCI/containment cooling system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

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3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the LPCI/containment cooling system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, containment cooling service water (CCSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. 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. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, 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. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

Valves and blind flanges located in high radiation areas may be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low. Normally locked or sealed closed penetrations may be opened intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the penetration. In

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this way, the penetration can be rapidly isolated when a valid secondary containment isolation signal is indicated.

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. 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. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. The emergency power supply OPERABILITY requirements for the standby gas treatment system are addressed within Specification 3.9.A, Actions. For example, if conducting the alternate offsite power source cross-

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tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

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3/4.8.A Containment Cooling Service Water System

The containment cooling service water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the containment cooling system and of other safety-related equipment (e.g., CCSW keep-fill, the control room emergency ventilation system refrigeration units), during normal and accident conditions. The redundant cooling capacity of the system, assuming a single failure, is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. Since only two of the four pumps is required to provide the necessary cooling capacity, a thirty day repair period is allowed for one pump out of service. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.B Diesel Generator Cooling Water System

The diesel generator cooling water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the diesel generators during normal and accident conditions. The cooling capacity of the system is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.C Ultimate Heat Sink

The canals provide an ultimate heat sink with sufficient cooling capacity to either provide normal cooldown of the units, or to mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown on the other unit.

3/4.8.D Control Room Emergency Ventilation System

The control room emergency filtration system maintains habitable conditions for operations personnel during and following all design basis accident conditions. This system, in conjunction with control room design, is based on limiting the radiation exposure to personnel occupying the room to five rem or less whole body, or its equivalent.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The control room emergency filtration system in-place testing procedures are established utilizing applicable sections of ANSI N510-1980 standard. Operation of the system with the heaters OPERABLE for ten hours a month is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The charcoal adsorber efficiency test procedures allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of ASTM-D-3803-89. The sample is at least two inches in diameter and has a length equivalent to the thickness of the bed. If the iodine removal efficiency test results are

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unacceptable, all adsorbent in the system is replaced. HEPA filter particulate removal efficiency is verified to be at least 99% by in-place testing with a DOP testing medium.

The control room refrigeration control unit (RCU) provides conditioned air for personnel comfort, safety and equipment reliability. The testing of the control room RCU system verifies that the heat-removal capability of the system is sufficient to remove sufficient heat load from the control room such that the control room air temperature is $\leq$ 95 °F. The test frequency is appropriate since significant degradation of the control room RCU system is not expected over this time period.

3/4.8.E Flood Protection

Flood protection measures are provided to protect the systems and equipment necessary for safe shutdown during high water conditions. The equipment necessary to implement the appropriate measures, as detailed in plant procedures, is required to be available, but not necessarily onsite, to implement the procedures in a timely manner. The selected water levels are based on providing timely protection from the design basis flood of the river.

3/4.8.F Snubbers

Mechanical snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic event or other event initiating dynamic loads. Snubbers are classified and grouped by design, manufacturer and accessibility. A list of individual snubbers with information of snubber location, classification or group, and system affected is maintained at the plant. The accessibility of each snubber is determined and documented for each snubber. The determination is based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operation (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to the systems. Therefore, the required inspection interval varies with the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly as determined and documented prior to the inspections. The categorization is used as the basis for determining the next inspection interval for that category.

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If a review and evaluation can not justify continued operation with an unacceptable snubber, the snubber is declared inoperable and the applicable action taken. To determine the next surveillance interval, the unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is OPERABLE in its as-found condition by the performance of a functional test. The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval, depending on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. The inspection interval may be as long as 48 months and the provisions of Specification 4.0.B may be applied.

When a snubber is found to be 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 adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide additional assurance of snubber functional reliability , a representative sample of the installed snubbers will be functionally tested at 18 month intervals. This sample is identified using one of three methods:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.8.F-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.8.F-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the NRC if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted are listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replace, 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 provide statistical bases for future consideration of snubber service life.

BASES

3/4.8.G 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, including startup sources and fission detectors, are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. 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 General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The release rates are determined at a more expeditious frequency following the determination of an increase of greater than 50%, as indicated by the air ejector noble gas monitor, after factoring out increases due to changes in THERMAL POWER level and off-gas flow in the nominal steady-state fission gas release from the primary coolant.

3/4.8.J Liquid Holdup Tanks

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 Appendix B, Table 2, Column 2, in unrestricted areas. Recirculation of the tank contents for the purpose of reducing the radioactive content is not considered to be an addition of radioactive material to the tank.

3.9 - LIMITING CONDITIONS FOR OPERATION

A. A.C. Sources - Operating

As a minimum, the following A.C. electrical power sources shall be OPERABLE:

1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

2. Two separate and independent diesel generators, each with:

   a. A separate fuel oil day tank containing ≥205 gallons of available fuel,

   b. A separate bulk fuel storage system containing ≥10,000 gallons of available fuel, and

   c. A separate fuel oil transfer pump.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required offsite circuit power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

A. A.C Sources - Operating

1. Each of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE:

   a. At least once per 7 days by verifying correct breaker alignments and indicated power availability, and

   b. At least once per 18 months by manually transferring the power supply from the normal circuit to the alternate circuit.

2. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 31 days by:

   a. Verifying the fuel levels in both the fuel oil day tank and the bulk fuel storage tank.

   b. Verifying the fuel transfer pump starts and transfers fuel from the bulk fuel storage system to the fuel oil day tank.

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a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Restore the inoperable offsite circuit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one of the above required diesel generator power sources inoperable:

    a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

    b. If the diesel generator is inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 24 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

4.9 - SURVEILLANCE REQUIREMENTS

c. Verifying[c] the diesel starts and accelerates to synchronous speed with generator voltage and frequency at 4160 ±420 volts and 60 ±1.2 Hz, respectively.

d. Verifying the diesel generator is synchronized, loaded to between 2470 and 2600 kW[d] in accordance with the manufacturer's/vendor's recommendations, and operates with this load for ≥60 minutes.

e. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

f. Verifying the pressure in required starting air receiver tanks to be ≥220 psig.

3. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 31 days and after each operation of the diesel where the period of operation was ≥1 hour by removing any accumulated water from the day tank.

4. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 92 days by checking for and removing accumulated water from the fuel oil bulk storage tanks.

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b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

c Surveillance Requirement 4.9.A.7 may be substituted for Surveillance Requirement 4.9.A.2.c.

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore the diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one of the above offsite circuit power sources and one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit power source by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than preplanned maintenance or testing, demonstrate the OPERABILITY[e] of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 8 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours for each OPERABLE diesel generator.

4.9 - SURVEILLANCE REQUIREMENTS

5. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling new fuel oil prior to addition to the storage tanks in accordance with applicable ASTM standards, and

   b. Verifying prior to addition to the storage tanks that the sample meets the applicable ASTM standards for API gravity, water and sediment, and the visual test for free water and particulate contamination, and

   c. Verifying within 31 days of obtaining the sample that the kinematic viscosity is within applicable ASTM limits.

6. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling and analyzing the bulk fuel storage tanks at least once per 31 days in accordance with applicable ASTM standards, and

   b. Verifying that the sample meets the applicable ASTM standards for water and sediment, kinematic viscosity, and ASTM particulate contaminant is <10 mg/liter.

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e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore at least one of the inoperable A.C. power 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, and

d. Restore both offsite circuits and both diesel generators to OPERABLE status within 7 days from the time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With one of the above required diesel generator power sources inoperable, in addition to ACTION 2 or 3, as applicable:

   a. Verify within 2 hours that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply.

   b. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

7. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 184 days by verifying[c] the diesel starts and accelerates to synchronous speed in $\leq$13 seconds. The generator voltage and frequency shall be verified to reach 4160 $\pm$420 volts and 60 $\pm$1.2 Hz, respectively, in $\leq$13 seconds after the start signal.

8. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 18 months by:

   a. Deleted.

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a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

c Surveillance Requirement 4.9.A.7 may be substituted for Surveillance Requirement 4.9.A.2.c.

3.9 - LIMITING CONDITIONS FOR OPERATION

5. With two of the above required offsite circuit power sources inoperable:

    a. Restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

    b. Restore at least two offsite circuits to OPERABLE status within 7 days from the 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.

6. With both of the above required diesel generator power sources inoperable:

    a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

b. Verifying the diesel generator capability to reject its largest single emergency load ($\geq$642 kW) while maintaining speed $\leq$1001 rpm and voltage at 4160 $\pm$ 420 volts.

c. Verifying the diesel generator capability to reject a load between 2470 and 2600 kw[d], without tripping on overspeed. The generator voltage shall not exceed 5000 volts during or following the load rejection.

d. Simulating a loss of offsite power by itself, and:

    1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

    2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in $\leq$13 seconds, energizes the auto-connected shutdown loads, and operates with this load for $\geq$5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 $\pm$420 volts and 60 $\pm$1.2 Hz, respectively, during this test.

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Within 2 hours, restore at least one of the above required diesel generators to OPERABLE[e] status and verify that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Demonstrate the continued OPERABILITY of the restored diesel generator by performing Surveillance Requirement 4.9.A.2.c within the subsequent 72 hours, and

d. Restore at least two required diesel generators to OPERABLE status within 7 days from the 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.

7. With the fuel oil contained in the bulk fuel storage tank(s) not meeting the properties specified in Surveillance Requirements 4.9.A.5 and 4.9.A.6, restore the fuel oil properties to within the specified limits within 7 days. Otherwise, declare the associated diesel generator(s) inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

e. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for ≥ 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz, respectively, in ≤ 13 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

f. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and

1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤ 13 seconds, energizes the auto-connected emergency loads through the load sequencer, and operates with this load for ≥ 5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz, respectively, during this test.

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e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

3.9 - LIMITING CONDITIONS FOR OPERATION | 4.9 - SURVEILLANCE REQUIREMENTS

g. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential current are automatically bypassed upon an emergency actuation signal.

h. Verifying the diesel generator operates for ≥24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to between 2730 and 2860 kW[d] and during the remaining 22 hours of this test, the diesel generator shall be loaded to between 2470 and 2600 kW[d]. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤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 4.9.A.2.c[f].

i. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2860 kW.

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d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

f If Surveillance Requirement 4.9.A.2.c is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at approximately full load for 2 hours or until the operating temperature has stabilized.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

j. Verifying the diesel generator's capability to:

   1) synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   2) transfer its loads to the offsite power source, and

   3) be restored to its standby status.

k. Verifying that the automatic load sequence logic is OPERABLE with the interval between each load block within ±10% of its design interval.

9. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, and verifying that both diesel generators accelerate to ≥900 rpm in ≤13 seconds.

10. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.

a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

TABLE 4.9.A-1

DIESEL GENERATOR TEST SCHEDULE

(NOT USED)

3.9 - LIMITING CONDITIONS FOR OPERATION

B. A.C. Sources - Shutdown

As a minimum, the following A.C. electrical power sources shall be OPERABLE:

1. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and
2. One diesel generator with:
    a. A fuel oil day tank containing ≥205 gallons of available fuel,
    b. A bulk fuel storage system containing ≥10,000 gallons of available fuel, and
    c. A fuel oil transfer pump.

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5, and when handling irradiated fuel in the secondary containment.

ACTION:

1. With less than the above required A.C. electrical power sources OPERABLE:
    a. Suspend CORE ALTERATIONS,
    b. Suspend handling of irradiated fuel in the secondary containment,
    c. Suspend operations with a potential for draining the reactor vessel, and

4.9 - SURVEILLANCE REQUIREMENTS

B. A.C Sources - Shutdown

Each of the required A.C. electrical power sources shall be demonstrated OPERABLE per the surveillance requirements in Specification 4.9.A, except for 4.9.A.2.d.

3.9 - LIMITING CONDITIONS FOR OPERATION

d. Suspend crane operations over the spent fuel storage pool if fuel assemblies are stored therein.

2 In addition, when in OPERATIONAL MODE 5 with the water level <23 feet above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

3. The provisions of Specification 3.0.C are not applicable.

4.9 - SURVEILLANCE REQUIREMENTS

3.9 - LIMITING CONDITIONS FOR OPERATION

C. D.C. Sources - Operating

As a minimum, the following D.C. electrical power sources shall be OPERABLE with the identified parameters within the limits specified in Table 4.9.C-1:

1. Two station 250 volt batteries, each with a full capacity charger.
2. Two station 125 volt batteries, each with a full capacity charger.
3. Two unit 24/48 volt batteries, each with a full capacity charger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required 24/48 volt or 250 volt station batteries and/or chargers inoperable, restore the inoperable equipment to OPERABLE status within 2 hours[b].

4.9 - SURVEILLANCE REQUIREMENTS

C. D.C. Sources - Operating

Each of the required 24/48 volt, 125 volt and 250 volt batteries and chargers shall be demonstrated OPERABLE[a]:

1. At least once per 7 days by verifying that:
   a. The parameters in Table 4.9.C-1 meet Category A limits, and
   b. There is correct breaker alignment to the battery chargers and total battery terminal voltage is ≥26.0, ≥125.9, or ≥260.4 volts, as applicable, on float charge.
2. At least once per 92 days and within 7 days after a battery discharge with a battery terminal voltage below 21.7, 105 or 210 volts, as applicable, or battery overcharge with battery terminal voltage above 30, 150 or 300 volts, as applicable, by verifying that:
   a. The parameters in Table 4.9.C-1 meet the Category B limits,
   b. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is ≤150 x10^{-6} ohms or ≤20% above baseline connection resistance, whichever is higher, and

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a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE, except the Unit 2 total battery terminal voltage on float charge shall be verified weekly as ≥130.2 volts.

b Each 250 volt battery may be inoperable for a maximum of seven days per operating cycle for maintenance or testing. If it is determined that a 250 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days per operating cycle.

3.9 - LIMITING CONDITIONS FOR OPERATION

2. With one of the above required 125 volt station batteries and/or chargers inoperable, within 2 hours[c], either restore the inoperable equipment to OPERABLE status, or place an OPERABLE corresponding alternate 125 volt battery (with an OPERABLE full capacity charger) in service.

3. With the provisions of either ACTION 1 or 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With any Category A parameter(s) outside the limit(s) shown in Table 4.9.C-1, the battery may be considered OPERABLE provided that its associated charger is OPERABLE, and 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.

5. With any Category B parameter(s) outside the limit(s) shown in Table 4.9.C-1, 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 the limit(s) within 7 days.

4.9 - SURVEILLANCE REQUIREMENTS

c. The average electrolyte temperature of all connected cells is above 60°F.

3. At least every 18 months by verifying that:

a. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

b. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

c. The resistance of each cell-to-cell and terminal connection is $\leq 150 \times 10^{-6}$ ohms or $\leq 20\%$ above baseline connection resistance, whichever is higher.

d. The battery chargers will supply a load equal to the manufacturer's rating for at least 4 hours.

4. At least every 18 months, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for design duty cycle when the battery is subjected to a battery service test.

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c With Unit 2 and 3 in OPERATIONAL MODE(s) 1, 2 or 3, each 125 volt battery may be inoperable for up to a maximum of seven days per operating cycle for maintenance or testing provided the alternate 125 volt battery is placed into service and is OPERABLE. If it is determined that a 125 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days provided the alternate 125 volt battery is placed into service and is OPERABLE. With the other Unit in MODE(s) 4 or 5, operations may continue with one of the two 125 volt battery systems inoperable provided the alternate 125 volt battery is placed into service and is OPERABLE.

3.9 - LIMITING CONDITIONS FOR OPERATION

6. With any Category B parameter not within its allowable value(s), immediately declare the battery inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

5. At least once per 60 months, verify that the battery capacity is 80% of the manufacturer's rating when subjected to either a performance discharge test or a modified performance discharge test. The modified performance discharge test satisfies the requirements of both the service test and performance test and therefore, may be performed in lieu of a service test.

6. For any battery that shows signs of degradation or has reached 85% of the service life for the expected application and delivers a capacity of less than 100% of the manufacturer's rated capacity, a performance discharge test or a modified performance test of battery capacity shall be performed at least once every 12 months or the battery shall be replaced or restored to 100% or greater of the manufacturer's rated capacity during the next refuel outage. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity and has shown no signs of degradation, a performance test or a modified performance test of battery capacity shall be performed at least once every two years.

TABLE 4.9.C-1

BATTERY SURVEILLANCE REQUIREMENTS

PARAMETER	CATEGORY A	CATEGORY B	
	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼" above maximum level indication mark	> Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	≥ 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195[b], and Average of all connected cells > 1.205[b]	Not more than 0.020 below the average of all connected cells, and Average of all connected cells ≥ 1.195[b]

TABLE NOTATIONS

(a) Corrected for electrolyte temperature and level.

(b) Or battery charging current is less than 2 amperes when on float charge.

(c) May be corrected for average electrolyte temperature.

3.9 - LIMITING CONDITIONS FOR OPERATION

D. D.C. Sources - Shutdown

As a minimum, the following D.C. electrical power sources shall be OPERABLE:

1. One station 250 volt battery with a full capacity charger.

2. One station 125 volt battery with a full capacity charger.

3. One unit 24/48 volt battery with a full capacity charger.

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5, and when handling irradiated fuel in the secondary containment.

ACTION:

With any of the above required station batteries and/or associated charger(s) inoperable, suspend CORE ALTERATIONS, suspend handling of irradiated fuel in the secondary containment, and suspend operations with a potential for draining the reactor vessel.

4.9 - SURVEILLANCE REQUIREMENTS

D. D.C. Sources - Shutdown

The required batteries and chargers shall be demonstrated OPERABLE[a] per the surveillance requirements in Specification 4.9.C.

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a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE, except the Unit 2 total battery terminal voltage on float charge shall be verified weekly as ≥ 130.2 volts.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 2, Nos. 23-1 and 24-1,

      2) For Unit 3, Nos. 33-1 and 34-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 2, Nos. 28 and 29,

      2) For Unit 3, Nos. 38 and 39.

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. For Unit 2, TB MCC 2 and RB MCC 2,

   b. For Unit 3, TB MCC 3 and RB MCC 3.

3. For Unit 2, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 2A-1and 3A,

   b. TB Res. Bus Nos. 2B and 2B-1,

   c. Reserve Bus No. 2, and

   d. RB Distribution Panel No. 2.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

4. For Unit 3, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 2A-1, 3A and 3A-1,

   b. TB Res. Bus Nos. 3B and 3B-1, and

   c. RB Distribution Panel No. 3.

5. 24/48 volt D.C. power distribution, consisting of:

   a. For Unit 2, Bus Nos. 2A and 2B.

   b. For Unit 3, Bus Nos. 3A and 3B.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTIONS:

1. With one of the above required A.C. distribution systems not energized, re-energize the system 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 one of the above required D.C. distribution systems not energized, re-energize the system within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3.9 - LIMITING CONDITIONS FOR OPERATION

F. Distribution - Shutdown

The following power distribution systems shall be energized with:

1. A.C. power distribution consisting of:
   a. One Unit engineered safety features 4160 volt bus:
      1) For Unit 2, No. 23-1 or 24-1,
      2) For Unit 3, No. 33-1 or 34-1.
   b. One associated Unit engineered safety features 480 volt bus:
      1) For Unit 2, No. 28 or 29,
      2) For Unit 3, No. 38 or 39.
2. For Unit 2, 125 volt D.C. power distribution, consisting of either:
   a. TB Main Bus No. 2A-1, and RB Distribution Panel No. 2, or
   b. TB Main Bus No. 3A, Reserve Bus No. 2, and TB Res. Bus Nos. 2B and 2B-1.
3. For Unit 3, 125 volt D.C. power distribution, consisting of either:
   a. TB Main Bus Nos. 3A and 3A-1, and RB Distribution Panel No. 3, or
   b. TB Main Bus No. 2A-1 and TB Res. Bus Nos. 3B and 3B-1.

4.9 - SURVEILLANCE REQUIREMENTS

F. Distribution - Shutdown

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

4. For 24/48 volt D.C. distribution, either:

    a. Bus Nos. 2A and 2B, or

    b. Bus Nos. 3A and 3B.

APPLICABILITY:

OPERATIONAL MODE(s) 4, 5, and when handling irradiated fuel in the secondary containment.

ACTIONS:

With less than the above required A.C. or D.C. distribution systems energized, suspend CORE ALTERATIONS, suspend handling of irradiated fuel in the secondary containment, and suspend operations with a potential for draining the reactor vessel.

4.9 - SURVEILLANCE REQUIREMENTS

3.9 - LIMITING CONDITIONS FOR OPERATION

G. RPS Power Monitoring

Two Reactor Protection System (RPS) electric power monitoring CHANNEL(s) for each inservice RPS Motor Generator (MG) set or alternate power supply shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4[a] and 5[a]

ACTION:

1. With one RPS electric power monitoring CHANNEL for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable power monitoring CHANNEL to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.
2. With both RPS electric power monitoring CHANNEL(s) for an inservice RPS MG set or alternate power supply inoperable, restore at least one electric power monitoring CHANNEL to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.9 - SURVEILLANCE REQUIREMENTS

G. RPS Power Monitoring

The specified RPS electric power monitoring CHANNEL(s) shall be determined OPERABLE:

1. By performance of a CHANNEL FUNCTIONAL TEST[b] each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.
2. 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:
   a. Overvoltage ≤129.6 volts AC
   b. Undervoltage ≥105.3 volts AC
   c. Underfrequency ≥55.4 Hz

a With any control rod withdrawn.

b Only required to be performed prior to entering MODE 2 or 3 from MODE 4.

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through a 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis. With suitable redundancy

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through a 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis. With suitable redundancy

BASES

in components and features not available, the plant must be placed in a condition for which the Limiting Condition for Operation does not apply.

The term verify as used toward A.C. electrical power sources means to administratively check by examining logs or other information to determine if certain components are out-of-service for pre-planned preventative maintenance, testing, or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

With one offsite circuit and one diesel generator inoperable, individual redundancy is lost in both the offsite and onsite electrical power system. Therefore, the allowable outage time is more limited. The time limit takes into account the capacity and capability of the remaining sources, reasonable time for repairs, and the low probability of a design basis event occurring during this period.

With both of the required offsite circuits inoperable, sufficient onsite A.C. sources are available to maintain the unit in a safe shutdown condition in the event of a design basis transient or accident. In fact, a simultaneous loss of offsite A.C. sources, a loss-of-coolant accident, and a worst-case single failure were postulated as a part of the design basis in the safety analysis. Thus, the allowable outage time provides a period of time to effect restoration of all or all but one of the offsite circuits commensurate with the importance of maintaining an A.C. electrical power system capable of meeting its design intent.

With two diesel generators inoperable there are no remaining standby A.C. sources. Thus, with an assumed loss of offsite electrical power, insufficient standby A.C. sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of A.C. power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown, which could result in grid instability and possibly a loss of total A.C. power. The allowable time to repair is severely restricted during this condition. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

Surveillance Requirements are provided which assure proper circuit continuity for the offsite A.C. electrical power supply to the onsite distribution network and availability of offsite A.C. electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure distribution buses and loads are connected to their preferred power source. The frequency is adequate since breaker position is not likely to change without the operator being aware of it and because status is displayed in the control room. Should the action provisions of this specification require an increase in frequency, this Surveillance Requirement assures proper circuit continuity for the available offsite A.C. sources during periods of degradation and potential information on common cause failures that would otherwise go undiscovered.

BASES

Surveillance Requirements are also provided for demonstrating the OPERABILITY of the diesel generators. The specified testing is based on the guidance provided in Regulatory Guide 1.9, Revision 3 (7/93), Regulatory Guide 1.108, Revision 1, and Regulatory Guide 1.137, Revision 1, as modified by plant specific analysis, diesel generator manufacturer/vendor recommendations and responses to Generic Letter 84-15.

The diesel generators are equipped with a prelubrication system which maintains a continuous flow of oil to the diesel engine moving parts while the engine is shutdown. The purpose of this system is to increase long term diesel generator reliability by reducing the stress and wear caused by frequent dry starting of the diesel generator. The diesel generator prelube may be accomplished either through normal operation of the installed prelubrication system or by manual prelubrication of the diesel generator in accordance with the manufacturer's/vendor's instructions. Performance of an idle start of the diesel generator is not considered to be a means of prelubrication.

A periodic "start test" of the diesel generators demonstrates proper startup from standby conditions, and verifies that the required generator voltage and frequency is attained. For this test, the diesel generator may be slow started and reach rated speed on a prescribed schedule that is selected to minimize stress and wear. In cases where this Surveillance Requirement is being used to identify a possible common mode failure in accordance with the action provisions, this test eliminates the risk of paralleling two of the remaining A.C. sources, which may compromise the A.C. source independence.

A "load-run test" normally follows the periodic "start test" of the diesel generator to demonstrate operation at or near the continuous rating. This surveillance should only be conducted on one diesel generator at a time in order to avoid common mode failures that might result from offsite circuit or grid perturbations. A minimum run time of 60 minutes is required to stabilize engine temperatures. Actual run time should be in accordance with vendor recommendations with regard to good operating practice and should be sufficient to ensure that cooling and lubrication are adequate for extended periods of operation, while minimizing the time that the diesel generator is connected to the offsite source. This Surveillance Requirement may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. A load band is provided to avoid routine overloading of the diesel generators. Momentary transients outside the load band because of changing bus loads do not impact the validity of this test.

A periodic surveillance requirement is provided to assure the diesel generator is aligned to provide standby power on demand. Periodic surveillance requirements also verify that, without the aid of the refill compressor, sufficient air start capacity for each diesel generator is available. With either pair of air receiver tanks at the minimum specified pressure, there is sufficient air in the tanks to start the associated diesel generator.

The periodicity of surveillance requirements for the shared diesel generators shall be equivalent to those required for the unit diesel generators. For example, it is not the intention to perform surveillances for the shared diesel generators twice during the specified surveillance interval in order to satisfy each unit's diesel generator surveillance requirements. By appropriately staggering

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the surveillance intervals between all three (3) diesel generators further ensures that for any loaded diesel generator surveillances, not more than one diesel generator is rendered inoperable at any given time in order to perform such testing.

Surveillance requirements provide verification that there is an adequate inventory of fuel oil in the storage tanks that is sufficient to provide time to place the facility in a safe shutdown condition and to bring in replenishment fuel from an offsite location. Additional diesel fuel can normally be obtained and delivered to the site within an eight hour period; thus a two day supply provides for adequate margin. The operation of each required fuel oil transfer pump is demonstrated by transferring fuel oil from its associated storage tank to its associated day tank. This surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the necessary fuel oil day tank instrumentation is OPERABLE.

A comprehensive surveillance program is provided to ensure the availability of high quality fuel oil for the diesel generators which is necessary to ensure proper operation. Water content should be minimized, because water in the fuel would contribute to excessive corrosion of the system, causing decreased reliability. The growth of micro-organisms results in slime formations, which are one of the chief causes of jellying in hydrocarbon fuels. Therefore, minimizing such slimes is also essential to assuring high reliability.

Sampling of both new diesel fuel oil and the bulk fuel oil storage tanks is in accordance with the American Society for Testing Materials (ASTM) standard D4057. Testing for API gravity is in accordance with ASTM D1298, water and sediment is in accordance with ASTM D1796, and the visual test for free water and particulate contamination (clear and bright) is in accordance with ASTM D4176. Testing for kinematic viscosity is in accordance with ASTM D445 and particulate contaminant testing is in accordance with ASTM D2276. Parameter limits are in accordance with ASTM D396 for API gravity, ASTM D975 for water and sediment and for kinematic viscosity, and ASTM D4176 for "clear and bright." The specific revision in use for each of these standards is controlled by procedure.

The diesel fuel oil day tanks are not equipped with the capability to obtain samples. Any accumulated water is removed by partially draining the day tank to the bulk fuel oil storage tank on a routine basis. Monthly sampling of the bulk fuel oil storage tank is then used to detect the presence of any water.

Fuel oil testing may indicate that such fuel oil is not within the required parameters. However, continued operation is acceptable while measures are taken to restore the properties of the fuel oil to within its limits since the properties of interest, even if they were not within the required limits, would not have an immediate effect on diesel generator operation. If the fuel oil properties cannot be returned to within their limits in the allowed time, the associated diesel generator(s) is (are) declared inoperable and the appropriate ACTION(s) taken.

A semi-annual surveillance is provided to verify the diesel generator can "fast start" from standby conditions and achieve the required voltage and frequency within the timing assumptions of the

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design basis loss of coolant accident safety analysis. Conducting this test on a semi-annual frequency is consistent with the intent of the reduction of cold testing identified in Generic Letter 84-15.

Additional surveillance requirements provide for periodic inspections and demonstration of the diesel generator capabilities, some are conducted in conjunction with a simulated loss of offsite power and/or a simulated ESF actuation signal. These tests of the diesel generator are expected to be conducted during an outage to functionally test the system. This testing is consistent with the intent of the diesel generator reliability programs recommended by Regulatory Guide 1.155.

3/4.9.B A.C. Sources - Shutdown

The A.C. sources required during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

2. Systems needed to mitigate a fuel handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

With one or more of the required A.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

The Surveillance Requirements for A.C. Source Shutdown are the same as those for operation, with the exception of the periodic "load-run test" which is not required due to the limited redundancy of A.C. power sources.

3/4.9.C D.C. Sources - Operating

The station D.C. electrical power system provides the A.C. emergency power system with control power. It also provides both motive and control power to selected safety-related equipment. During normal operation, the D.C. electrical loads are powered from the battery chargers with batteries floating on the system. In case of loss of normal power to the battery charger, the D.C. load is automatically powered from the station batteries.

BASES

Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source.

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

BASES

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each connection provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The limits established for this Surveillance Requirement shall be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

Verifying an acceptable average temperature of battery cells is consistent with the recommendations of IEEE-450 and ensures that lower than normal temperatures do not act to inhibit or reduce battery capacity.

Verifying that the chargers will provide the manufacturer's rated current and voltage for four hours ensures that charger deterioration has not occurred and that the charger will provide the necessary capacity to restore the battery to a fully charged state.

A battery service test is a special test of the battery's capability "as found" to satisfy the design requirements of the D.C. electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

A battery modified performance test is a test of the battery capacity and the battery's ability to meet the loads that exceed the constant current discharge rate of the battery (high rate short duration loads) of the battery's duty cycle. This test satisfies the requirements of both a service test and a performance test and is intended to detect any change in capacity and to determine overall battery degradation due to age and usage. The batteries have a rated capacity of 125% of the load expected at the end of their service life allowing for a minimum battery capacity of at least 80% of the manufacturer's rating. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

3/4.9.D D.C. Sources - Shutdown

The D.C. sources required to be OPERABLE during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent drain down of the reactor vessel;

2. Systems needed to mitigate a fuel-handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE;

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

BASES

With one or more of the required D.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

3/4.9.E Distribution - Operating

The OPERABILITY of the A.C. and D.C. onsite power distribution systems ensures that sufficient power will be available to 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 surveillance requirements verify that the A.C. and D.C. electrical power distribution systems are functioning properly, with all the required circuit breakers closed and the buses energized from normal power. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The frequency takes into account the redundant capability of the A.C. and D.C. electrical power distribution subsystems, and other indications available in the control room that will alert the operator to subsystem malfunctions.

3/4.9.F Distribution - Shutdown

The OPERABILITY of the minimum specified A.C. and D.C. onsite power distribution systems, during Cold Shutdown and Refueling and when handling irradiated fuel in the secondary containment, ensures that the facility can be maintained in these conditions for extended time periods and sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. Requiring OPERABILITY of the minimum specified onsite power distribution systems when handling irradiated fuel in the secondary containment helps to ensure that systems needed to mitigate a fuel handling accident are available.

3/4.9.G RPS Power Monitoring

Specifications are provided to ensure the OPERABILITY of the reactor protection system (RPS) bus electrical protection assemblies (EPAs). Each RPS motor generator (MG) set and the alternate power source has 2 EPA CHANNEL(s) wired in series. A trip of either CHANNEL from either overvoltage, undervoltage, or underfrequency will disconnect the associated MG set or alternate power source.

The associated surveillance requirements provide for demonstration of the OPERABILITY of the RPS EPA's. The setpoints for overvoltage, undervoltage, and underfrequency have been chosen based on analysis (ref. February 4, 1983 letter to H. Denton from T. Rausch).

3.10 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Mode Switch

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:

1. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

2. CORE ALTERATION(s) 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.

   a. All rods in.
   b. Refuel platform position.
   c. Refuel platform hoists fuel-loaded.
   d. Fuel grapple position.

APPLICABILITY:

OPERATIONAL MODE 5[a][b].

ACTION:

1. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATION(s) and lock the reactor mode switch in the Shutdown or Refuel position.

4.10 - SURVEILLANCE REQUIREMENTS

A. Reactor Mode Switch

1. The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified:

   a. Within 2 hours prior to:

      1. Beginning CORE ALTERATION(s), and

      2. Resuming CORE ALTERATION(s) when the reactor mode switch has been unlocked.

   b. At least once per 12 hours.

2. Each of the required reactor mode switch Refuel position interlocks[c] shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST within 24 hours prior to the start of and at least once per 7 days during control rod withdrawal or CORE ALTERATION(s), as applicable.

3. Each of the required reactor mode switch Refuel position interlocks[c] that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or

---

a See Special Test Exceptions 3.12.A and 3.12.B

b The reactor shall be maintained in OPERATIONAL MODE 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

c 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 individual.

3.10 - LIMITING CONDITIONS FOR OPERATION

2. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

3. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATION(s) with equipment associated with the inoperable Refuel position equipment interlock.

4.10 - SURVEILLANCE REQUIREMENTS

CORE ALTERATION(s), as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

3.10 - LIMITING CONDITIONS FOR OPERATION

B. Instrumentation

At least 2 source range monitor[a] (SRM) CHANNEL(s) shall be OPERABLE and inserted to the normal operating level with:

1. Continuous visual indication in the control room,
2. One of the required SRM detectors located in the quadrant where CORE ALTERATION(s) are being performed and the other required SRM detector located in an adjacent quadrant, and
3. Unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3.3.A and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.10.A, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn[b].

APPLICABILITY:

OPERATIONAL MODE 5, unless the following conditions are met:

1. No more than two fuel assemblies are present in each core quadrant associated with an SRM;

4.10 - SURVEILLANCE REQUIREMENTS

B. Instrumentation

Each of the required SRM channels shall be demonstrated OPERABLE by:

1. At least once per 12 hours:
   a. Performance of a CHANNEL CHECK.
   b. Verifying the detectors are inserted to the normal operating level, and
   c. During CORE ALTERATION(s), verifying that the detector of an OPERABLE SRM CHANNEL is located in the core quadrant where CORE ALTERATION(s) are being performed and another is located in an adjacent quadrant.
2. Performance of a CHANNEL FUNCTIONAL TEST:
   a. Within 24 hours prior to the start of CORE ALTERATION(s), and
   b. At least once per 7 days.
3. Verifying that the channel count rate is at least 3 cps:
   a. Prior to control rod withdrawal,
   b. Prior to and at least once per 12 hours during CORE ALTERATION(s),
   c. At least once per 24 hours.

---

a The use of special movable detectors during CORE ALTERATION(s) in place of the normal SRM neutron detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

b Not required for control rods removed per Specification 3.10.I and 3.10.J

3.10 - LIMITING CONDITIONS FOR OPERATION

2. While in the core, these two fuel assemblies are in locations adjacent to the SRM; and

3. In the case of movable detectors, each group of fuel assemblies shall be separated by at least two fuel cell locations from any other fuel assemblies.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATION(s) and fully insert all insertable control rods.

4.10 - SURVEILLANCE REQUIREMENTS

4. Verifying, within 8 hours prior to and at least once per 12 hours during the time any control rod is withdrawn[b] that the "shorting links" have been removed from the RPS circuitry unless adequate SHUTDOWN MARGIN has been demonstrated per Specification 3.3.A and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.10.A.

b Not required for control rods removed per Specification 3.10.I or 3.10.J

3.10 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod Position

All control rods shall be fully inserted[a].

APPLICABILITY:

OPERATIONAL MODE 5 during CORE ALTERATION(s)[b].

ACTION:

With all control rods not fully inserted, suspend all other CORE ALTERATION(s), except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

4.10 - SURVEILLANCE REQUIREMENTS

C. Control Rod Position

All control rods shall be verified to be fully inserted, except as specified:

1. Within 2 hours prior to:
   a. The start of CORE ALTERATION(s).
   b. The withdrawal of one control rod under the control of the reactor mode switch Refuel position one-rod-out interlock.
2. At least once per 12 hours.

---

a Except control rods removed per Specification 3.10.I or 3.10.J or one control rod withdrawn under control of the reactor mode switch refuel position one-rod-out interlock.

b See Special Test Exception 3.12.B

3.10 - LIMITING CONDITIONS FOR OPERATION

D. Deleted

4.10 - SURVEILLANCE REQUIREMENTS

D. Deleted

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3.10 - LIMITING CONDITIONS FOR OPERATION

E. Communications

Direct communication shall be maintained between the control room and refueling platform personnel.

APPLICABILITY:

OPERATIONAL MODE 5, during CORE ALTERATION(s)[a].

ACTION:

When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATION(s).

4.10 - SURVEILLANCE REQUIREMENTS

E. Communications

Direct communication between the control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATION(s).

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a Except movement of control rods with their normal drive system.

3.10 - LIMITING CONDITIONS FOR OPERATION

F. DELETED

4.10 - SURVEILLANCE REQUIREMENTS

F. DELETED

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3.10 - LIMITING CONDITIONS FOR OPERATION

G. Water Level - Reactor Vessel

At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY:

During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL MODE 5 when the fuel assemblies or control rods being handled are irradiated or the fuel assemblies or control rods 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 condition109

4.10 - SURVEILLANCE REQUIREMENTS

G. Water Level - Reactor Vessel

The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of ≥33 feet.

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.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

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.

3.10 - LIMITING CONDITIONS FOR OPERATION

I. Single Control Rod Removal

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.

1. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1-2 and Specification 3.10.A.
2. The source range monitors (SRM) are OPERABLE per Specification 3.10.B.
3. The SHUTDOWN MARGIN requirements of Specification 3.3.A are satisfied, except that the control rod selected to be removed;
   a. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and
   b. Need not be assumed to be immovable or unscrammable.
4. All other control rods in a five-by-five array centered on the control rod being removed are either:
   a. Fully inserted and electrically or hydraulically disarmed, or
   b. 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.

4.10 - SURVEILLANCE REQUIREMENTS

I. Single Control Rod Removal

Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control drive mechanism are reinstalled and the control rod is fully inserted in the core, verify that:

1. The reactor mode switch is OPERABLE per Surveillance Requirement 4.1.A.1 or 4.10.A.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 3.10.A.
2. The SRM CHANNEL(s) are OPERABLE per Specification 3.10.B.
3. The SHUTDOWN MARGIN requirements of Specification 3.3.A are satisfied per Specification 3.10.I.3.
4. All other control rods in a five-by-five array centered on the control rod being removed are either:
   a. Fully inserted and electrically or hydraulically disarmed, or
   b. 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.
5. All other control rods are fully inserted.

3.10 - LIMITING CONDITIONS FOR OPERATION

5. All other control rods are fully inserted.

APPLICABILITY:

OPERATIONAL MODE(s) 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.

4.10 - SURVEILLANCE REQUIREMENTS

3.10 - LIMITING CONDITIONS FOR OPERATION

J. Multiple Control Rod Removal

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 fully inserted in the core.

1. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.10.A, 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.

2. The source range monitors (SRM) are OPERABLE per Specification 3.10.B.

3. The SHUTDOWN MARGIN requirements of Specification 3.3.A are satisfied.

4. All other control rods are either fully inserted or have the surrounding four fuel assemblies removed from the core cell.

5. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

APPLICABILITY:

OPERATIONAL MODE 5.

4.10 - SURVEILLANCE REQUIREMENTS

J. Multiple Control Rod Removal

1. Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are fully inserted in the core, verify that:

   a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.1.A.1 or 4.10.A.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.10.A.

   b. The SRM CHANNEL(s) are OPERABLE per Specification 3.10.B.

   c. The SHUTDOWN MARGIN requirements of Specification 3.3.A are satisfied.

   d. All other control rods are either fully 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.

3.10 - LIMITING CONDITIONS FOR OPERATION

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.

4.10 - SURVEILLANCE REQUIREMENTS

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.

3.10 - LIMITING CONDITIONS FOR OPERATION

K. Shutdown Cooling and Coolant Circulation - High Water Level

At least one shutdown cooling (SDC) loop shall be OPERABLE and in operation[a], with at least:

1. One OPERABLE SDC pump, and

2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 5, when irradiated fuel is in the reactor vessel and the water level is $\geq$23 feet above the top of the reactor pressure vessel flange.

ACTION:

1. With no SDC loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

2. With no SDC loop in operation, within one hour establish reactor coolant circulation by an alternate method, monitor reactor coolant temperature at least once per hour, and verify reactor coolant circulation at least once per 12 hours.

4.10 - SURVEILLANCE REQUIREMENTS

K. Shutdown Cooling and Coolant Circulation - High Water Level

At least one SDC loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

a The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

3.10 - LIMITING CONDITIONS FOR OPERATION

L. Shutdown Cooling and Coolant Circulation - Low Water Level

Two shutdown cooling (SDC) loops shall be OPERABLE and at least one loop shall be in operation[a], with each loop consisting of at least:

1. One OPERABLE SDC pump, and
2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 5, when irradiated fuel is in the reactor vessel and the water level is < 23 feet above the top of the reactor pressure vessel flange.

ACTION:

1. With less than the above required SDC loops OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the OPERABILITY of at least one alternate method capable of decay heat removal for each inoperable SDC loop.
2. With no SDC loop in operation, within one hour establish reactor coolant circulation by an alternate method, monitor reactor coolant temperature at least once per hour, and verify reactor coolant circulation at least once per 12 hours.

4.10 - SURVEILLANCE REQUIREMENTS

L. Shutdown Cooling and Coolant Circulation - Low Water Level

At least one SDC loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

a The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

BASES

3/4.10.A 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.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the Refuel position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. If the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

3/4.10.B 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, whenever reactor criticality is possible.

The source range monitors (SRM) are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and reactor startup. Requiring two OPERABLE source range monitors in and adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. Requiring a minimum of 3 counts per second whenever criticality is possible provides assurance that neutron flux is being monitored. The SRM system is designed to provide a signal-to-noise ratio of at least 3:1 and a count rate of at least 3 counts per second. Criticality is considered to be impossible if there are no more than two assemblies in a quadrant and if these are in locations adjacent to the source range monitors (i.e., spatially separated).

Special movable detectors may be used during CORE ALTERATION(s) in place of the normal SRM neutron detectors. These special detectors must be connected to the normal SRM circuits such that the applicable neutron flux indication, control rod blocks and scram signals can be generated. The special detectors provide more flexibility in monitoring reactivity changes during fuel loading since they can be positioned anywhere within the core during refueling provided they meet the location requirements of the specification.

When the Reactor Protection System shorting links are removed, the source range monitors provide added protection against local criticalities by providing an initiating signal for a reactor scram on high neutron flux.

BASES

3/4.10.C Control Rod Position

The requirement that all control rods be inserted during other CORE ALTERATION(s) ensures that fuel will not be loaded into a cell without an inserted control rod.

3/4.10.D DELETED

3/4.10.E Communications

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status regarding core reactivity conditions during movement of fuel within the reactor pressure vessel.

3/4.10.F DELETED

3/4.10.G Water Level - Reactor Vessel

3/4.10.H Water Level - Spent Fuel Storage Pool

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

BASES

3/4.10.I Single Control Rod Removal

3/4.10.J Multiple 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.

Refueling interlocks restrict the movement of control rods and the operation of the refueling equipment to reinforce operational procedures that prevent the reactor from becoming critical. These interlocks prevent the withdrawal of more than one control rod. Under these conditions, since only one control rod can be withdrawn, the reactor core will always be shut down even with the highest worth control rod withdrawn if adequate SHUTDOWN MARGIN exists. Verification that all the other control rods are fully inserted is required to assure the SHUTDOWN MARGIN is within the limits. Verification that the five-by-five array of control rods are inserted and disarmed while the scram function for the withdrawn control rod is not available is required to ensure that the possibility of criticality remains precluded.

During refueling operations, no more than one control rod is permitted to be withdrawn from a core cell containing one or more fuel assemblies. When all four fuel assemblies are removed from a core cell, the control rod may be withdrawn with no restrictions. With no fuel assemblies in the core cell, the associated control rod has no reactivity control function and is not required to remain inserted. Prior to reloading fuel into the core cell, the associated control rod must be inserted to ensure that an inadvertent criticality does not occur.

3/4.10.K Shutdown Cooling and Coolant Circulation - High Water Level

3/4.10.L Shutdown Cooling and Coolant Circulation - Low Water Level

The requirement that at least one shutdown cooling loop be OPERABLE and in operation or that an alternate method capable of decay heat removal be demonstrated ensures that 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.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating loop will not result in a complete loss of shutdown cooling capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange a large heat sink is available for core cooling. Thus, in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3.11 - LIMITING CONDITIONS FOR OPERATION

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGR) for each type of fuel as a function of bundle average exposure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective action within 15 minutes, and
2. Restore APLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

A. AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

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 APLHGR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. TRANSIENT LINEAR HEAT GENERATION RATE

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be maintained such that the FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) is less than or equal to 1.0. Where FDLRC is equal to:

$$\frac{(LHGR)\ (1.2)}{(TLHGR)\ (FRTP)}$$

APPLICABILITY:

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

ACTION:

With FDLRC greater than 1.0, initiate corrective ACTION within 15 minutes and within 6 hours either:

1. Restore FDLRC to less than or equal to 1.0, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC, or
3. Adjust[a] each APRM gain such that the APRM readings are ≥100% times the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

B. TRANSIENT LINEAR HEAT GENERATION RATE

The value of FDLRC shall be verified:

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 FDLRC greater than or equal to 1.0.
4. The provisions of Specification 4.0.D are not applicable.

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a Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

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.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

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

ACTION:

With an LHGR exceeding the SLHGR limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the LHGR to within the SLHGR limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

The SLHGR shall be determined to be equal to or less than the limit:

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 SLHGR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

E. DELETED

4.11 - SURVEILLANCE REQUIREMENTS

E. DELETED

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BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification 6.9.

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B TRANSIENT LINEAR HEAT GENERATION RATE

The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that ≥1% plastic strain does not occur; and, the fuel does not experience centerline melt during anticipated operational occurrences beginning at any power level and terminating at 120% of RATED THERMAL POWER. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

The daily requirement for calculating FDLRC when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when

BASES

there have not been significant power or control rod changes. The requirement to calculate FDLRC 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 FDLRC after initially determining FDLRC is greater than 1.0 exists to ensure that FDLRC will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

The FUEL DESIGN LIMIT RATIO FOR CENTERLINE MELT (FDLRC) is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)} ;$$

where LHGR is the LINEAR HEAT GENERATION RATE, and TLHGR is the TRANSIENT LINEAR HEAT GENERATION RATE. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C 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 the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in Specification 6.9.

The purpose of the reduced flow MCPR curves specified in the CORE OPERATING LIMITS REPORT are to define MCPR operating limits at other than rated core flow conditions. The reduced flow MCPR curves assure that the Safety Limit MCPR will not be violated.

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 has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for

BASES

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 after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

This specification assures that the maximum LINEAR HEAT GENERATION RATE in any fuel rod is less than the design STEADY STATE LINEAR HEAT GENERATION RATE even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distributions 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 SLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that SLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

3.12 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

The provisions of Specifications 3.7.A, 3.7.E and 3.10.A 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 212°F.

APPLICABILITY:

OPERATIONAL MODE 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 212°F, immediately place the reactor mode switch in the Shutdown position.

4.12 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

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.

3.12 - LIMITING CONDITIONS FOR OPERATION

B. SHUTDOWN MARGIN Demonstrations

The provisions of Specifications 3.10.A and 3.10.C 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.

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B.
2. The rod worth minimizer is OPERABLE per Specification 3.3.L 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 individual.
3. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.
4. No other CORE ALTERATION(s) are in progress.

APPLICABILITY:

OPERATIONAL MODE 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.

4.12 - SURVEILLANCE REQUIREMENTS

B. SHUTDOWN MARGIN Demonstrations

Within 30 minutes prior to and at least once per 12 hours during the performance of a SHUTDOWN MARGIN demonstration, verify that;

1. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.10.B,
2. The rod worth minimizer is OPERABLE with the required program per Specification 3.3.L or a second licensed operator or other technically qualified individual is present and verifies compliance with the SHUTDOWN MARGIN demonstration procedures, and
3. No other CORE ALTERATION(s) are in progress.

BASES

3/4.12.A 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. Low power PHYSICS TESTS during OPERATIONAL MODE 2 may be required to be performed while still maintaining access to the primary containment and reactor pressure vessel. Additional requirements during these tests to restrict reactor power and reactor coolant temperature provide protection against potential conditions which could require primary containment or reactor coolant pressure boundary integrity.

3/4.12.B SHUTDOWN MARGIN Demonstrations

Performance of SHUTDOWN MARGIN demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO. SHUTDOWN MARGIN tests may be performed while in OPERATIONAL MODE 2 in accordance with Table 1-2 without meeting this Special Test Exception. For SHUTDOWN MARGIN demonstrations performed while in OPERATIONAL MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM, or must be verified by a second licensed operator or other technically qualified individual. To provide additional protection against inadvertent criticality, control rod withdrawals that are "out-of-sequence", i.e., do not conform to the Banked Position Withdrawal Sequence, must be made in individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATION(s) may be in progress. This Special Test Exception then allows changing the Table 1-2 reactor mode switch position requirements to include the Startup or Hot Standby position such that the SHUTDOWN MARGIN demonstrations may be performed while in OPERATIONAL MODE 5.

5.0 DESIGN FEATURES

5.1 SITE

Site and Exclusion Area

5.1.A The site consists of approximately 953 acres adjacent to the Illinois River at the point where it is formed by the confluence of the Des Plaines and Kankakee Rivers, in the northeast quarter of the Goose Lake Township, Grundy County, Illinois. The Exclusion Area shall not be less than 800 meters from the centerline of the reactor vessels.

Low Population Zone

5.1.B The Low Population Zone shall be a five mile radius from the centerline of the reactor vessels.

Radioactive Gaseous Effluents

5.1.C Information regarding radioactive gaseous effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

Radioactive Liquid Effluents

5.1.D Information regarding radioactive liquid effluents shall be located in the OFFSITE DOSE CALCULATION MANUAL.

FIGURE 5.1.A-1

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FIGURE 5.1.B-1

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5.0 DESIGN FEATURES

5.2 CONTAINMENT

Configuration

5.2.A The primary containment is a steel lined concrete structure consisting of a drywell and suppression chamber. The drywell is a steel structure composed of a spherical lower portion, a cylindrical middle portion, and a hemispherical top head. The drywell is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 158,236 cubic feet. The suppression chamber has an air region of 116,300 to 112,800 cubic feet and a water region of 116,300 to 119,800 cubic feet.

Design Temperature and Pressure

5.2.B The primary containment is designed and shall be maintained for:

1.	Maximum internal pressure:	62 psig.
2.	Maximum internal temperature:	drywell 281°F. suppression pool 281°F.
3.	Maximum external pressure:	drywell 2 psig. suppression pool 1 psig.

Secondary Containment

5.2.C The secondary containment consists of the Reactor Building and a portion of the main steam tunnel and has a minimum free volume of 5,760,000 cubic feet.

5.0 DESIGN FEATURES

5.3 REACTOR CORE

Fuel Assemblies

5.3.A The reactor core shall contain 724 fuel assemblies. Each assembly consists of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material and water rods. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Control Rod Assemblies

5.3.B The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.0 DESIGN FEATURES

5.4 REACTOR COOLANT SYSTEM

Design Pressure and Temperature

5.4.A The reactor coolant system is designed and shall be maintained:

1. In accordance with the code requirements specified in Section 5 of the UFSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

2. For a pressure and temperature of:

    a. 1175 psig at 565°F on the suction side of the recirculation pump.

    b. 1450 psig at 575°F from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

    c. 1325 psig at 580°F from the discharge shutoff valve to the jet pumps.

Volume

5.4.B The total water and steam volume of the reactor vessel and recirculation system is approximately 14,626 cubic feet at 68°F.

5.0 DESIGN FEATURES

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5.6 FUEL STORAGE

Criticality

5.6.A The spent fuel storage racks are designed and shall be maintained with:

1. A k_{eff} equivalent to ≤ 0.95 when flooded with unborated water, including all calculational uncertainties and biases as described in Section 9.1 of the UFSAR.

2. A nominal 6.30 inch center-to-center distance between fuel assemblies placed in the storage racks.

Drainage

5.6.B The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 589′ 2.5".

Capacity

5.6.C The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3537 fuel assemblies.

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.A The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.B The Shift Manager shall be responsible for directing and commanding the safe overall operation of the facility under all conditions.

6.2 ORGANIZATION

6.2.A Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Quality Assurance Manual.

2. The Station Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

3. The Chief Nuclear Officer (CNO) shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.B Unit Staff

The unit staff shall include the following:

1. Three non-licensed operators shall be on site at all times.

2. At least one licensed Reactor Operator shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE(s) 1, 2, 3 or 4 at least one licensed Senior Reactor Operator shall be present in the control room.

3. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.B.1 and 6.2.C for a period of time not to exceed two 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.

4. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than two hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g, senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

   The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

6. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

6.2.C Shift Technical Advisor

The Shift Technical Advisor (STA) shall provide technical advisory support to the Unit Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the facility. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A single STA may fulfill this function for both units.

6.3 UNIT STAFF QUALIFICATIONS

Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971, "Selection and Training of Nuclear Plant Personnel", dated March 8, 1971, except for the Radiation Protection Manager, who shall meet or exceed the qualifications of the Radiation Protection Manager as specified in Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

A retraining and replacement program for the unit staff shall be maintained under the direction of the appropriate on site manager. Training shall be in accordance with ANSI N18.1-1971 and 10 CFR 55 for appropriate designated positions and shall include familiarization with relevant industry operational experience.

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6.7 SAFETY LIMIT VIOLATION

6.7.A The following actions shall be taken in the event a Safety Limit is violated:

1. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Site Vice-President or his designated alternate shall be notified within 24 hours;

2. Within 30 days, a Licensee Event Report (LER) shall be prepared documenting the event pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC.

3. Critical operation of the Unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

1. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,
2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,
3. Station Security Plan implementation,
4. Generating Station Emergency Response Plan implementation,
5. PROCESS CONTROL PROGRAM (PCP) implementation,
6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and
7. Fire Protection Program implementation.

6.8.B Deleted.

6.8.C Deleted.

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, IC, process sampling (post accident sampling of reactor coolant and containment atmosphere), containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

2. In-Plant Radiation Monitoring

   This program provides controls which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

   a. Training of personnel,

   b. Procedures for monitoring, and

   c. Provisions for maintenance of sampling and analysis equipment.

3. Post Accident Sampling

   This program provides controls which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and primary containment atmosphere samples under accident conditions. The program shall include the following:

   a. Training of personnel,

   b. Procedures for sampling and analysis,

   c. Provisions for maintenance of sampling and analysis equipment.

4. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by station procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

b. Limitations on the instantaneous concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to ten (10) times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2 to 10 CFR Part 20.1001 - 20.2402,

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

d. Limitations on the annual and quarterly doses to a member of the public from radioactive materials in liquid effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

f. Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose conforming to Appendix I to 10 CFR Part 50,

g. Limitations on the dose rate resulting from radioactive materials released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

a) For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ.

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each Unit to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50,

i. Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with halflives greater than 8 days in gaseous effluents released from each Unit conforming to Appendix I to 10 CFR Part 50,

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 1.6% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the combined Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

b. Air lock testing acceptance criteria is the overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq$ P_a.

The provisions of 4.0.B do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of 4.0.C are applicable to the Primary Containment Leakage Rate Testing Program.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the appropriate Regional Office of the NRC unless otherwise noted.

6.9.A. Routine Reports

1. Deleted

2. Annual Report

   Annual reports covering the activities of the Unit for the previous calendar year, as described in this section shall be submitted prior to May 1 of each year.

   The reports required shall include:

   a. Tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated person rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter or TLD. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

   b. The results of specific activity analysis in which the reactor coolant exceeded the limits of Specification 3.6.J. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to 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 reactor coolant exceeded the radioiodine limit.

3. Annual Radiological Environmental Operating Report

   The Annual Radiological Environmental Operating Report covering the operation of the Unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

4. Radioactive Effluent Release Report

   The Radioactive Effluent Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to April 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

5. Monthly Operating Report

   Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety valves or safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

6. CORE OPERATING LIMITS REPORT

   a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

      (1) The Control Rod Withdrawal Block Instrumentation for Table 3.2.E-1 of Specification 3.2.E.

      (2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit for Specification 3.11.A.

      (3) The Steady State Linear Heat Generation Rate (SLHGR) for Specification 3.11.D.

      (4) The Minimum Critical Power Operating Limit (including 20% scram insertion time) for Specification 3.11.C. This includes rated and off-rated flow conditions.

b. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of topical reports:

(1) ANF-1125(P)(A), "Critical Power Correlation - ANFB."

(2) ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

(3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

(4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

(5) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

(6) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology."

(7) ANF-913(P)(A), "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis."

(8) Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods", and associated Supplements on Neutronics Licensing Analyses (Supplement 1) and La Salle County Unit 2 Benchmarking (Supplement 2).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.B Special Reports

Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

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6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.A Pursuant to 10 CFR 20.1601(c), in lieu of the requirements of paragraph 20.1601 of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm (12 in.) 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)[a] (or equivalent document). Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

1. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

2. 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; or

3. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the RWP (or equivalent document).

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a Health Physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

6.12.B In addition to the requirements of 6.12.A, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP(or equivalent document).

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. Deleted.

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.A Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

    a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

    b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after review and acceptance, including approval by the Station Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.A Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

   b. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance, including approval by the Station Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change to the ODCM was made effective. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.